Beaver Valley in which the SE cited RIS 2005-29 and yet relied on the EPRI testing data to address the concern.

2.10 2005 SECY-05-0138

In August 2005, the NRC staff issued SECY-05-0138, "Risk-Informed and Performance-Based Alternatives to the Single-Failure Criterion." SECY-05-0138 presents a comprehensive history of the application of the single failure criterion including extensive discussion of the treatment of passive components in fluid systems. The paper acknowledges that "[o]ne particular issue identified in this project is the continued existence of the footnote to the definition of single failure in 10 CFR 50 Appendix A stating that the regulatory position on considering passive failures in fluid systems is under development." The paper quotes from SECY-77-439 (discussed above) and recognizes that in current practice, as in 1977, ~~that~~ "[p]assive failures in fluid systems are generally excluded from single-failure assessments."

SECY-05-0138 presents three alternatives for using a risk-informed and performance-based approach to address the single failure issue. The paper states that "[a]ll alternatives could include developing a position on single passive failures in fluid systems to replace the footnote now in 10 CFR Part 50 Appendix A definitions."

The paper also makes it clear that, with few exceptions, neither the NRC staff nor the Commission has established specific requirements relating to the treatment of passive component failures in fluid systems. The Panel believes the existence of this Commission paper, contemporaneous with discussions on potential safety valve failures (e.g.~~-~~., ~~RIS~~ RIS 2005-~~029~~-29), makes it clear that no specific "established standards" on safety valve failures had been developed between 1977 and the time of the Byron and Braidwood ~~power uprate decisions~~license amendments in 2001 and 2004.

Standard Review Plan (SRP) Section 15.5.1 (2007)

SRP Section 15.5.1 (2007) states, "The pressurizer safety valves, too, may be assumed to reseat properly after having relieved water; but only if such valves have been qualified for water relief." However, this section does not reference ASME requirements for qualification.

Comment [SW]: Check for first use of SRP. I think it was above, but it may not have been defined.

2.12 NRR 2015 Compliance Backfit Finding and the Appeals

The NRR 2015 compliance backfit (October 9, 2015, letter to Exelon) is predicated on the following positions ~~(emphases added)~~:

- "water relief through a valve that is not qualified for water relief will cause that valve to stick in its fully open position" [emphasis added]
- "the licensee ... has not applied the single-failure assumption" [emphasis added]
- "nor have they provided ASME water qualification documentation for the PSVs ... the ASME ... original Overpressure Protection Report ... inservice test history ... including both water and steam tests" [emphasis added]

The compliance backfit ~~then~~ argues that the IOECCS ~~(an AOO per the GDC definition and a Condition II event in the ANS classification system adopted in the Byron and Braidwood UFSARs)~~ would escalate to a more severe event. Such an escalation would be contrary to the Byron and Braidwood licensing basis (i.e., contrary to the ANS non-escalation position) and could be in non-compliance with the GDCs (as included in the Byron and Braidwood licensing basis) since an IOECCS with a stuck-open valve had not been analyzed and shown to meet to the appropriate criteria for an AOO.

~~The Panel has reviewed the October 9, 2015, backfit letter to Exelon, the Exelon response and appeal letter of December 8, 2015 (referred to as the NRR appeal), and multiple documents associated with the NRR appeal, including:~~

~~the Exelon meeting presentation of March 7, 2016 in defense of their appeal~~

~~the NRR appeal panel meeting minutes (January 15, 2016; February 5, 2016; February 25, 2016; and March 15, 2016)~~

~~a memorandum documenting the input of one NRR appeal panel member (A Gody to M Bailey March 21, 2016)~~

~~the May 3, 2016, appeal response by the director of NRR, including the NRR appeal panel's report~~

~~As noted previously, Exelon appealed to the NRC EDO, and NEI provided a letter supporting this appeal; the Panel reviewed these documents as well.~~

Based on ~~a~~ its review of all the relevant documents, and discussions with ~~numerous parties~~ the individuals (staff and managers) involved in the original review and the compliance backfit ~~proposal~~, the Panel has developed a good understanding of the regulatory requirements and practices, the potential safety issues, and backfit rule obligations. The Panel has determined that the numerous, complex, and detailed regulatory and technical issues all depend on the answers to ~~three~~ two critical questions on valve performance, namely:

- Must the PSVs in question be assumed to fail given liquid water discharge because of the lack of ASME certification for water discharge?
- ~~May~~ Must the PSVs be assumed to fail in accordance with the GDC "single failure" requirements?

~~Are the Byron or Braidwood PSVs so likely to fail and to stick open, given liquid water discharge, that the NRC staff must assume their failure as a normal consequence of that condition and not as an "independent fault," but as a consequential failure?~~

In the~~ir~~ October 9, 2015, letter to Exelon, the NRC staff indicates that "[o]ne assumption that is particularly important to the non-escalation criteria is that water relief through a valve that is not qualified for water relief will cause that valve to stick in its fully open position" [emphasis added]. The Panel concludes that this issue, the treatment of potential valve failure, is not only "particularly important," it is the critical issue upon which the compliance backfit hinges.

3 RESPONSE TO THE EDO QUESTIONS

In establishing the Panel, the EDO asked the panel to answer five specific questions, as well as evaluating the overall appropriateness of the October 2015 backfit. The answers to these questions are provided below.

Comment [CT]: Per Michael's comment – should confirm.

Comment [CT]: Proposing to delete based on Michael's comment – I think I agree that the first question covers it.

Comment [SW]: Please make conforming changes here for any of my edits accepted for the cover letter.

3.1 Were the approvals based on a mistake? If so, what was the mistake and what are the implications for Braidwood and Byron?

In responding the question, the Panel has considered the differing views of the NRR staff and the licensee on this issue. Those positions are summarized below:

- In the May 3, 2016, letter to Exelon, the NRC staff claims that "[t]he NRC erred in approving a sequence of events that allowed the [IOECCS], [CVCS] malfunction, and inadvertent opening of a pressurizer safety or relief valve analyses in the 2001 and 2004 [SEs]" and "the NRC staff understood the PSVs to be qualified for water relief when, in fact, they were not."

- Exelon claims, in ~~their~~ the December 8, 2015, backfit appeal letter, that "the compliance exception requires more than simply asserting that the prior staff approvals were wrong—the NRC must demonstrate that the prior approvals were erroneous because of an omission or mistake of fact at the time of the approval. The NRC has not made that case here."

The Panel concludes that, in 2001 and 2004, the NRC staff did not misunderstand the qualification status of the PSVs and that it was not a ~~categorical~~ mistake to undertake a review of or make a technically based safety finding on the likely successful performance of the valves. In the Panel's opinion, the actions of the Reactor Systems Branch in 2001 to reach out to the Division of Engineering/Mechanical Engineering Branch for expert technical review assistance was both appropriate and ~~noteworthy~~commendable. The NRC staff reviewers involved in the 2001 power uprate review were among the most experienced and senior reviewers in their areas of expertise. The NRC staff valve expert involved in the review was the agency's most knowledgeable individual on PSVs and the relevant ASME Code requirements, and was a nationally recognized expert. The Panel cannot agree that the NRC staff was misinformed, ill-informed, or in error, or that ~~they~~ it made incorrect or inappropriate decisions.

Comment [CT]: Note when doing references – Michael recommends placeholder titles (e.g., NRR backfit appeal ruling).

Comment [CT]: Note for Michael – I think we typically don't hyphenate adverbs (which was your suggestion).

3.2 What is the known and established standard for water qualification of PSVs?

The Panel concludes that in 2001 and 2004 and at present, the known and established standard of the Commission is that the failures of ~~mechanical~~ passive ~~s~~Safety ~~v~~Valves need not be assumed to occur following water discharge ~~-~~ if the likelihood is sufficiently small, ~~—~~ based on well-informed staff engineering judgment~~—~~. No more detailed or prescriptive standard has been promulgated by the Commission.

3.3 What is the known and established standard for progression of postulated events between categories of severity?

For Byron and Braidwood, the standard for progression of postulated events between categories of severity is set forth in the UFSAR as described above. . The Panel supports the staff's view that non-escalation (from ANS category II to ANS category IV) is a known and established standard applicable to Byron and Braidwood. However, this event progression standard does not establish specific standards for valve qualification. Therefore, it is not the basis for a compliance backfit given this set of facts.

In answering this question, the Panel was also asked to include a discussion of RIS 2005-~~029~~29 and the draft Revision 1 to RIS 2005-29 that was issued for public comment in 2015.

The Panel has reviewed the issue of "event escalation" as discussed in the compliance backfit and in RIS 2005-~~029~~-29 and the draft Revision 1 to RIS 2005-29. The Panel concludes that the IOECCS (an AOO per the GDC definition and a Condition II event in the ANS classification system adopted in the Byron and Braidwood UFSARs) would escalate to a more severe event if a PSV were to stick open, or if both a PORV stuck open and its block valve failed to close. Such an escalation would be contrary to the Byron and Braidwood licensing basis (i.e., contrary to the ANS non-escalation position) and could be in non-compliance with the GDC (as included in the Byron and Braidwood licensing basis), since an IOECCS with a stuck-open valve had not been analyzed and shown to meet to appropriate criteria for an AOO.

Comment [CT]: Michael's comment: I'm not seeing the connection to the RIS? This seems more related to answering the first part of the EDO's question. Does the RIS really change anything here? Is this really just based on the UFSAR?

Should we delete/revise/etc.?

3.4 Does the current licensing basis for Braidwood and Byron comply with the applicable regulations? Is it adequate to provide protection to public health and safety?

The Panel concludes that Braidwood and Byron do comply with the applicable regulations based on the UFSAR analyses, which the NRC staff found acceptable through a reasonable and technically sound evaluation using appropriate Commission safety standards.

3.5 Given that Exelon suggests that the NRC pursue a cost-justified substantial safety enhancement backfit, what is the contribution to overall plant risk of the current configuration at Braidwood and Byron?

The Panel requested the Office of Nuclear Regulatory Research to provide information and insights on the risk significance of the sequence at issue, to assure that the Panel's judgments were being made with a full understanding of their significance, and to assist in responding to the EDO question.

The RES study suggests that the most significant IOECCS sequence, assuming that all pressurizer overfill events lead to a small LOCA, contributes approximately 1 percent of the total internal event core damage frequency (CDF). In ~~the~~ its report, RES estimated the maximum benefit (CDF reduction) from a "perfect backfit" (i.e., always preventing pressurizer overfill) ~~is estimated~~ at 1.5E-07 per year. If the PSVs are not assumed to always fail following water discharge (consistent with the staff expert judgment in 2001) or a smaller improvement than a "perfect backfit" were considered, the risk-reduction benefit of implementing the backfit w~~c~~ould be even smaller.

The Panel is aware of and sensitive to two important issues related to this question. First, NRR, not the appeal Panel, is responsible for any decisions on alternative application of the backfit rule to this issue (through the other categories of adequate protection or cost-justified substantial safety enhancement). Second, the Panel does not wish to imply that "the contribution to plant risk" should be seen as the only measure of enhanced safety. The issues of event classification and the non-escalation of events are essentially defense-in-depth concepts. Defense in depth has a recognized role and value in the regulatory process. The Panel is also aware that not every defense-in-depth feature has the same safety significance, and that the estimated risk significance (measured in core damage frequency) is very relevant.

Within the context ~~suggested~~ described above, the Panel concludes that the contribution to overall plant risk is very small [illegible].

4 SUMMARY AND CONCLUSIONS

The compliance exception to the Backfit Rule is intended to address failures to meet known and established Commission standards because of omission or mistake of fact. New or modified interpretations of what constitutes compliance do not fall within the exception. Therefore, to address the appeal of the proposed compliance backfit, the Panel has focused on determining if this case is most appropriately characterized as one in which the ~~NRC staff and the~~ licensee "failed to meet known and established standards of the Commission because of omission or mistake of fact," or rather as a case of a "new or modified interpretations of what constitutes compliance."

The NRC staff's compliance backfit argument depends on two separate determinations: the assumed failure of PSVs to reclose after passing water and the necessity of preventing "event escalation" (i.e., the position that "an incident of moderate frequency should not generate a more serious plant condition without other faults occurring independently"). For the staff's compliance backfit conclusion to be valid, both of these determinations must meet the above compliance backfit standard by involving failure to meet known and established standards of the Commission.

In the first of these determinations, the NRC staff's compliance backfit finding is based on the assumption in the SE that the PSV fails to reclose given the absence of "ASME water qualification documentation." ~~The compliance backfit asserts that staff was mistaken in using expert technical judgment and in reviewing the licensee's submittal documenting the EPRI valve test results (performed in response to an NRC requirement (TMI Action Item II.D.1) and previously evaluated and found acceptable by the NRC staff).~~

As indicated in the compliance backfit ~~proposal~~, the 2001 NRC staff SE for the Byron and Braidwood power uprate did involve a technical evaluation of safety valve capability and likely performance under water-discharge conditions rather than a simple assumption of a failure. The NRC response to the Exelon first appeal indicates that "the 2001 and 2004 approvals occurred because the NRC staff understood the PSVs to be qualified for water relief when, in fact, they were not."

The Panel has carefully considered these views and has reviewed the relevant documents including the NRC staff's RAIs and the licensee's responses, the NRC staff SE input, and the final staff SE written at the time of the 2001 power uprate review. 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 certification sense; rather, the record shows thorough consideration of the testing conducted on valves of the type installed at the plant and a technical judgment that this testing provided appropriate qualification.

~~The~~ On the basis of its review, the Panel concludes that the NRC staff who conducted the 2001 power uprate review did not misunderstand the qualification status of the PSVs and that it was not a ~~categorical~~ mistake to undertake a review of or make a technically based safety finding on the likely successful performance of the valves. In the Panel's opinion, the actions of the Reactor Systems Branch in 2001 to reach out to the Division of Engineering/Mechanical Engineering Branch for expert technical review assistance was both appropriate and ~~noteworthy~~commendable. The NRC staff reviewers involved in the 2001 power uprate review were among the most experienced and senior reviewers in their areas of expertise. The NRC staff valve expert involved in the review was the agency's most knowledgeable individual on PSVs and the relevant ASME Code requirements, and was a nationally recognized expert. The

Comment [CT]: Placeholder to add further discussion about new idea that the 2001 reviewer thought it had previously been approved to discharge water in this particular event (which it hadn't).

Comment [SW]: I like noteworthy better than commendable, but can live with either.

Panel cannot agree that the NRC staff was misinformed, ill-informed, or in error, or that ~~they~~ it made incorrect or inappropriate decisions.

In the Panel's opinion, three related technical and regulatory positions underpin the backfit:

- ASME water qualification (certification) documentation is required if a valve is to be assumed to reclose after passing water.
- ~~W~~water relief through a steam-qualified valve will cause that valve to stick in its fully open position.
- PSVs are subject to a single-failure assumption.

None of these positions were "known and established standards of the Commission" in 2001 or 2004 for determining when it was appropriate to assume a failure of PSVs to reseat. In fact, they were not "known and established standards of the Commission" in 2005 (RIS 2005-29) or 2006 (Beaver Valley EPU) or 2007 ~~when similar decisions were made~~(SRP Section 15.1.1 update).

Moreover, two of these positions do not appear to be "established standards of the Commission" at present. The NRC staff's call for ASME certification first appears in the Exelon compliance backfit. The call for ~~use of~~ applying the single failure criterion first appears in the proposed 2015 ~~Draft~~ draft Revision 1 to RIS 2005-~~029~~-29.

The Panel concludes that the standard in place in 2001 and 2004 and at present is simply that the failures of ~~mechanical~~ passive ~~S~~safety ~~V~~valves need not be assumed to occur following water discharge if the likelihood is sufficiently small, —based on well-informed staff engineering judgment. In earlier documents addressing this topic, beginning with NUREG-0737, ~~The standard is also that~~ the use of the word "qualified" or "qualification" implies ~~only~~ a general demonstration of capability, such as in the EPRI testing done in response to TMI Action Plan Item II.D.1. In light of ~~these~~ this standards, it is the Panel's opinion, that during the Exelon power uprate review in 2001, and the later review of a ~~later~~ valve setpoint amendment in 2004, the NRC staff exercised reasonable and well-informed engineering judgment when concluding that the PSVs were unlikely to stick open (i.e., fail to reseat).

~~The~~ Overall, the Panel has concluded that the NRC staff's position on valve qualification in the 2015 backfit is a new or modified interpretation of what constitutes compliance in addressing potential PSV failures following water discharge. Although this new staff position represents a well-intentioned and conservative approach that could provide additional safety margin, it does not provide a basis for a compliance backfit.

~~In~~ Finally, the absence of a PSV failure to reseat, the panel concluded that the concerns articulated by the staff in ~~the~~ its backfit 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 Panel's findings, therefore, support the Exelon backfit appeal.

5 ADDITIONAL PANEL THOUGHTS

In addition to the specific finding relating to the backfit appeal, the Panel believes it is important to acknowledge, and for the NRC staff and licensee to appreciate, that water discharge through an PSV not specifically designed for such service is undesirable and should be minimized or avoided as a matter of conservative engineering and prudent operations. The Panel concludes

Comment [CT]: Added per Michael's comment – confirm this is what was meant.

Comment [SW]: We use present and past tense. I'd like past tense throughout.

this while fully aware that the event sequence being considered appears to be of little safety significance. Nonetheless, operator training and emergency procedures to terminate the event before pressurizer filling, as well as the use of PORVs rather than relying solely on PSVs, are clearly preferred prudent measures, whether they form the facilities' UFSAR licensing basis and are assumed in the accident analyses or not.

Comment [CT]: Did not yet include Tom's paragraph as it merits more discussion and perhaps engagement with Exelon.

APPENDIX A: HISTORY OF THE BACKFIT RULE AND THE COMPLIANCE EXCEPTION

Comment [SW]: No comments on Appendix A. Looks good.

The Backfit Rule

Title 10 of the *Code of Federal Regulations* (10 CFR), Section 50.109, "Backfitting," was originally promulgated in 1970 (Volume 35 of the *Federal Register* (FR), page 5317). Because of perceived deficiencies in the rule, the U.S. Nuclear Regulatory Commission (NRC) substantially revised it in 1985 (50 FR 38097). The 1985 rule was challenged in court, and the U.S. Circuit Court for the District of Columbia (D.C. Circuit) vacated this rule in its entirety. The D.C. Circuit took this action because it concluded that the revised rule could be interpreted to allow the NRC to consider costs in defining or redefining what is required for adequate protection of the public health and safety. *Union of Concerned Scientists v. U.S. Nuclear Regulatory Com'n*, 824 F.2d 108, 119-20 (1987). In response, the NRC revised the Backfit Rule in 1988 (53 FR 20603) to remove any implication that costs could be considered in defining or redefining adequate protection. The 1988 revisions only differed from the 1985 rule to the extent necessary to address the court's concerns. The 1988 rule was also challenged in court, but this time the D.C. Circuit upheld the rule. *Union of Concerned Scientists v. U.S. Nuclear Regulatory Com'n*, 880 F.2d 552 (1989).

In its current form, 10 CFR 50.109(a)(1) defines backfitting as~~"~~

> ... the modification of or addition to systems, structures, components, or design of a facility; or the design approval or manufacturing license for a facility; or the procedures or organization required to design, construct or operate a facility; any of which may result from a new or amended provision in the Commission's regulations or the imposition of a regulatory staff position interpreting the Commission's regulations that is either new or different from a previously applicable staff position ... .

~~... ."~~Unless one of three specified exceptions apply, the NRC may impose a backfit only if it performs a backfit analysis in accordance with 10 CFR 50.109(a)(2) and determines in accordance with 10 CFR 50.109(a)(3) "that there is a substantial increase in the overall protection of the public health and safety or the common defense and security to be derived from the backfit and that the direct and indirect costs of implementation for that facility are justified in view of this increased protection."

Section 50.109(a)(4) sets forth the three exceptions to the requirements of 10 CFR 50.109(a)(2) and (a)(3). The first exception, the compliance exception, applies if the "modification is necessary to bring a facility into compliance with a license or the rules or orders of the Commission, or into conformance with written commitments by the licensee." 10 CFR 50.109(a)(4)(i). The second and third exceptions relate to actions ensuring adequate protection or to actions that involve defining or redefining adequate protection. 10 CFR 50.109(a)(4)(ii)-(iii).

Commission Policy

The Commission addressed its intended application of the compliance exception in the 1985 rulemaking (50 FR at 38103):

> The compliance exception is intended to address situations in which the licensee has failed to meet known and established standards of the Commission because

of omission or mistake of fact. It should be noted that new or modified interpretations of what constitutes compliance would not fall within the exception and would require a backfit analysis and application of the standard.

In the 1985 rule, the Commission acknowledged that staff interpretations of regulations are not legally binding, but the Commission also stated that "staff interpretations of broadly stated rules are often necessary to give a rule effect and in some instances may be a causal factor in initiating a backfit." *Id.* at 38102. The Commission also stated, "Many of the most important changes in plant design, construction, operation, organization, and training have been put in place at a level of detail that is expressed in staff guidance documents which interpret the intent of broad, generally worked [sic] regulations." *Id.* at 38103.[4]

Backfitting Guidance

Extensive information regarding the appropriate implementation of backfitting is provided in the NRC's 1990 Backfitting Guidelines (NUREG-1409). Relevant excerpts from this guidance are provided below.

Applicable Regulatory Staff Positions

According to NUREG-1409, t~~T~~o be a backfit, "a new or revised staff position or requirement must be involved, that is, there must be a change in content or applicability of the previously applicable regulatory staff position (in the direction of increased safety requirements) . . . ." ~~(at 3)~~ An applicable regulatory staff position is a requirement or position already specifically imposed on or committed to by a licensee. Examples of applicable regulatory staff positions include:

- ~~A requirement or position already specifically imposed on or committed to by a licensee is called an applicable regulatory staff position. There are several different types of positions, such as~~

- legal requirements, as in explicit regulations, orders, and plant licenses and in amendments, conditions, and technical specifications

- written licensee commitments such as those contained in the final safety analysis report, licensee event reports, and docketed correspondence, including responses to NRC bulletins, generic letters, inspection reports, or notices of violation and confirmatory action letters

- NRC staff positions that are documented explicit interpretations of more general regulations and are contained in documents such as the Standard Review Plan, branch technical positions, regulatory guides, generic letters, and bulletins

~~For the purpose of this report, a change in the applicable regulatory staff position will be subsequently referred to as a new or revised position.~~

~~This manual chapter is included as~~A similar list of examples is provided in Inspection Manual Chapter 0514 (1988), which is included as Appendix D to NUREG-1409. The manual chapter was referenced in the 1988 rulemaking, and a working draft was provided to the Commission in

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[4] The 1988 rulemaking neither revised the compliance exception as stated in the 1985 rule nor provided additional guidance on its interpretation.

SECY-88-102 for information. The manual chapter provides a definition of "applicable regulatory staff positions" that is slightly more detailed than the definition in NUREG-1409. This definition is quoted below, with additional detail beyond the NUREG-1409 emphasized in underlined text.

> Applicable regulatory staff positions are those already specifically imposed upon or committed to by a licensee at the time of the identification of a plant-specific backfit, and are of several different types and sources:
>
> a. Legal requirements such as in explicit regulations, orders, plant licenses (amendments, conditions, technical specifications). Note that some regulations have update features built in, as for example, 10 CFR 50.55a, Codes and Standards. Such update requirements are applicable as described in the regulation.
>
> b. Written commitments such as contained in the [Final Safety Analysis Report], [Licensee Event Reports], and docketed correspondence, including responses to Bulletins, responses to Generic Letters, Confirmatory Action Letters, responses to Inspection Reports, or responses to Notices of Violation.
>
> c. NRC staff positions[5] that are documented, approved, explicit interpretations of the more general regulations, and are contained in documents such as the [Standard Review Plan], Branch Technical Positions, Regulatory Guides, Generic Letters, and Bulletins; and to which a licensee or an applicant has previously committed to or relied upon. Positions contained in these documents are not considered applicable staff positions to the extent that staff has, in a previous licensing or inspection action, tacitly or explicitly excepted the licensee from part or all of the position.[6]

---

[5] Requirements may be imposed by rule or order. Staff interpretations such as examples of acceptable ways to meet requirements are not requirements in and of themselves.
[6] Imposition of a staff position from which a licensee has previously been excepted is a backfit.

How Regulatory Positions are Established

NUREG-1409 provides responses to a number of questions regarding backfitting. The following response was given to questions asking, "Is it appropriate for the NRC staff to rely on informal or formal communications to other licensees as official NRC positions? What about NRC tacit approval of documents?"

> Informal or formal communications to one licensee are not official positions to all licensees. Section 053 of Manual Chapter 0514 identifies what can be applied as official staff positions in a plant-specific context. They are legal requirements such as contained in explicit regulations, orders, and plant licenses; written commitments such as contained in final safety analysis reports, licenses event reports, and docketed correspondence; and documented, approved explicit interpretations such as contained in the [Standard Review Plan], branch technical positions, regulatory guides, generic letters, and bulletins. Orders, licenses, and written commitments are applicable only to a particular licensee.
>
> If the NRC staff previously exempted a licensee from a legal requirement or approved position, it is not applicable to that licensee for the purpose of backfit consideration. Explicit exemption would be done formally in writing. The Appendix to NRC Manual Chapter 0514 discusses tacit approval under reanalysis of issues. Two situations are covered. In the first case, staff review of a previously accepted licensee action or program may result in a requested change. This would be classified as a backfit because it represents a change in a previous staff position and would require a backfit analysis (or a documented evaluation if it meets one of the exceptions listed in the backfit rule). In the second case, a licensee submittal committing to a specific course of action that has not received timely NRC staff review is implemented by the licensee. In this case, it is considered that the NRC staff tacitly accepted the licensee's action since timely notice to the contrary was not given. If the NRC staff subsequently adopts a different position and requests a change in the licensee action, this change may be classified as a backfit and thus require a backfit analysis (or a documented evaluation if it meets one of the exceptions listed in the backfit rule).

NUREG-1409 also addresses a question regarding tacit approvals by an inspector: "If an inspector has previously accepted (i.e., provided tacit approval of) a licensee's method, does a specific request for change constitute a backfit and if so, is a backfit analysis required?" The response is:

> Cases where an inspector provides tacit approval are relatively rare. Simply not challenging a licensee's practice normally would not be considered tacit approval. The only example provided in Manual Chapter 0514 is a case where the NRC has indicated tacit approval by not acting in a reasonable time on a licensee submittal and the licensee has moved ahead to implement the proposal described in the submittal. For the purpose of this question, it would most likely arise in connection with review of a licensee response to an inspection report.
>
> Explicit approval could be provided in an inspection report that states that a particular approach is acceptable. However, conclusions of that nature are usually made in SEs rather than inspection reports.

Compliance Backfit Guidance

~~The Backfitting Guidelines~~NUREG-1409 gives the following response to ~~include~~ the question, ~~(on page 12)~~ "[h]ow does the backfit rule apply to new staff positions that reflect an evolving understanding of technical issues?" ~~The response is:~~

> ~~New or revised staff positions are backfits when they are imposed on licensees and result in a change in structures, systems, design, or procedures (as described in 10 CFR 50.109). A backfit analysis is required whenever new or revised positions are imposed to achieve cost-justified substantial safety enhancements. A backfit analysis is not required if the new or changed position is imposed to bring a facility into compliance or if it is necessary to provide assurance of adequate protection. In those cases, however, a written evaluation is needed to provide the objectives of and reasons for the modification and the basis for invoking the exception.~~
>
> An evolving understanding of issues does not, by itself, define which category fits a particular backfit. Judgment must be applied to the facts of each particular case to determine whether the backfit is for compliance, to provide adequate protection, to redefine adequate protection, or to achieve a cost-justified substantial safety enhancement. For example, with regard to compliance, the 1985 statement of considerations for 10 CFR 50.109 indicates that "the compliance exception is intended to address situations where the licensee has failed to meet known and established standards of the Commission because of omission or mistake of fact....new or modified interpretations of what constitutes compliance would not fall within the exception...."

NUREG-1409 also provides an example where an evolving understanding of technical issues resulted in a compliance backfit that was apparently justified for at least some licensees. ~~The Backfitting Guidelines further ask (on page 13) if it is "appropriate for the NRC staff to rely on informal or formal communications to other licensees as official NRC positions? What about NRC tacit approval of documents?" The response is:~~

> ~~Informal or formal communications to one licensee are not official positions to all licensees. Section 053 of Manual Chapter 0514 identifies what can be applied as official staff positions in a plant-specific context. They are legal requirements such as contained in explicit regulations, orders, and plant licenses; written commitments such as contained in final safety analysis reports, licenses event reports, and docketed correspondence; and documented, approved explicit interpretations such as contained in the [Standard Review Plan], branch technical positions, regulatory guides, generic letters, and bulletins. Orders, licenses, and written commitments are applicable only to a particular licensee.~~
>
> ~~If the NRC staff previously exempted a licensee from a legal requirement or approved position, it is not applicable to that licensee for the purpose of backfit consideration. Explicit exemption would be done formally in writing. The Appendix to NRC Manual Chapter 0514 discusses tacit approval under reanalysis of issues. Two situations are covered. In the first case, staff review of a previously accepted licensee action or program may result in a requested change. This would be classified as a backfit because it represents a change in a previous staff position and would require a backfit analysis (or a documented evaluation if it meets one of the exceptions listed in the backfit rule). In the~~

> ~~second case, a licensee submittal committing to a specific course of action that has not received timely NRC staff review is implemented by the licensee. In this case, it is considered that the NRC staff tacitly accepted the licensee's action since timely notice to the contrary was not given. If the NRC staff subsequently adopts a different position and requests a change in the licensee action, this change may be classified as a backfit and thus require a backfit analysis (or a documented evaluation if it meets one of the exceptions listed in the backfit rule).~~

~~The Backfitting Guidelines also consider an example in which the~~In response to industry claims that Bulletin 88-11 ~~lacks~~ lacked any backfitting justification, the staff responded~~. The response is~~:

> Although the justification was not printed in the bulletin, NRC Bulletin 88-11, "Pressurizer Surge Line Thermal Stratification," was justified as a backfit. It is an example of a backfit that was determined by the responsible NRC official to be required as a matter of compliance with existing requirements and commitments. The CRGR reviewed the bulletin and concurred. The regulations currently require licensees to meet the applicable codes of the American Society of Mechanical Engineers (ASME), *Boiler and Pressure Vessel Code.* Because of the NRC staff's concern with the integrity of the surge line, licensees were requested to perform their fatigue analysis in accordance with the latest ASME Section III requirements that incorporate high cycle fatigue analysis. The justification provided by the NRC staff was that previously unconsidered thermal stratification phenomenon may invalidate the existing analysis performed to confirm the integrity of the surge line.
>
> Subsequently, it was understood that some licensees believed that the NRC staff's rationale was in error because they were not committed to the latest ASME Section III requirements by virtue of their license commitment. However, the issue became moot because these licensees undertook the analysis voluntarily in view of the safety importance of the issue and the fact that previous versions of the ASME Code did not completely address the concern.

~~Finally, the Backfitting Guidelines (on page 15) pose the question that "[i]f an inspector has previously accepted (i.e., provided tacit approval of) a licensee's method, does a specific request for change constitute a backfit and if so, is a backfit analysis required?" The response is:~~

> ~~Cases where an inspector provides tacit approval are relatively rare. Simply not challenging a licensee's practice normally would not be considered tacit approval. The only example provided in Manual Chapter 0514 is a case where the NRC has indicated tacit approval by not acting in a reasonable time on a licensee submittal and the licensee has moved ahead to implement the proposal described in the submittal. For the purpose of this question, it would most likely arise in connection with review of a licensee response to an inspection report.~~
>
> ~~Explicit approval could be provided in an inspection report that states that a particular approach is acceptable. However, conclusions of that nature are usually made in SEs rather than inspection reports.~~

NRC ~~Manual Chapter 0514 (1988)~~

This manual chapter is included as Appendix D to NUREG-1409. The manual chapter was referenced in the 1988 rulemaking, and a working draft was provided to the Commission in SECY-88-102 for information. The manual chapter provides a definition of "applicable regulatory staff positions" that is slightly more detailed than the definition in NUREG-1409. This definition is quoted below, with additional detail beyond the NUREG-1409 emphasized in underlined text.

> Applicable regulatory staff positions are those already specifically imposed upon or committed to by a licensee at the time of the identification of a plant-specific backfit, and are of several different types and sources:
>
> a. Legal requirements such as in explicit regulations, orders, plant licenses (amendments, conditions, technical specifications). Note that some regulations have update features built in, as for example, 10 CFR 50.55a, Codes and Standards. Such update requirements are applicable as described in the regulation.
>
> b. Written commitments such as contained in the [Final Safety Analysis Report], [Licensee Event Reports], and docketed correspondence, including responses to Bulletins, responses to Generic Letters, Confirmatory Action Letters, responses to Inspection Reports, or responses to Notices of Violation.
>
> c. NRC staff positions[7] that are documented, approved, explicit interpretations of the more general regulations, and are contained in documents such as the [Standard Review Plan], Branch Technical Positions, Regulatory Guides, Generic Letters, and Bulletins; and to which a licensee or an applicant has previously committed to or relied upon. Positions contained in these documents are not considered applicable staff positions to the extent that staff has, in a previous licensing or inspection action, tacitly or explicitly excepted the licensee from part or all of the position.[8]

~~[7] Requirements may be imposed by rule or order. Staff interpretations such as examples of acceptable ways to meet requirements are not requirements in and of themselves.~~

~~[8] Imposition of a staff position from which a licensee has previously been excepted is a backfit.~~

APPENDIX B: QUALIFICATION OF PRESSURE RELIEF VALVES IN NUCLEAR POWER PLANTS IN RESPONSE TO TMI-2 ACCIDENT

Nuclear power plants in the United States use various types of pressure relief valves to protect personnel and equipment from overpressure events within reactor fluid systems. Pressure relief valves include safety valves, safety relief valves, and relief valves, with different designs, operating conditions, and requirements. The American Society of Mechanical Engineers (ASME) *Boiler and Pressure Vessel Code* (BPV Code), Section III, Division 1, specifies requirements for the design, operation, installation, and testing of pressure relief valves used for various functions in nuclear power plants. For example, the ASME BPV Code (2007 Edition) in Article NB-7000, Overpressure Protection, specifies requirements for steam and air or gas service for safety valves; steam, air or gas, and liquid service for safety relief valves; liquid service for relief valves; and steam, air or gas, and liquid service for pilot operated or power actuated pressure relief valves. The ASME *Code for Operation and Maintenance of Nuclear Power Plants* (OM Code) provides requirements for the preservice and inservice testing (IST) programs for pressure relief valves in nuclear power plants.

Braidwood, Units 1 and 2 (Braidwood) and Byron, Units 1 and 2 (Byron) are Westinghouse-designed pressurized-water reactors (PWRs) that received their construction permits under Title 10 of the *Code of Federal Regulations* (10 CFR), Part 50, in December 1975. Each pressurizer in these four reactor units is equipped with three pressurizer safety valves (PSVs) and two power-operated relief valves (PORVs). The three PSVs are Crosby Model HP-BP-86, size 6M6 (6-inch), spring loaded pop type opened by direct fluid pressure. The PORVs are Copes-Vulcan Model D-100-160 3-inch pneumatic-actuated globe valves that respond to a signal from the pressure sensing system or to manual control. Each PORV can be isolated by a motor-operated block valve.

The ASME BPV Code of record for the PSVs at Braidwood and Byron was the 1971 Edition through the Winter 1972 addenda of the ASME BPV Code, Section III. At the time of the Braidwood and Byron operating license review, NRC Standard Review Plan (SRP), Revision 1 (July 1981), Chapter 15.5.1-15.5.2, "Inadvertent Operation of ECCS and Chemical and Volume Control System Malfunction that Increases Reactor Coolant Inventory," and Chapter 15.6.1, "Inadvertent Opening of a PWR Pressurizer Pressure Relief Valve or a BWR [boiling-water reactor] Pressure Relief Valve," provided general staff guidance for these plant transients. In March 2007, the NRC staff issued Revision 2 to these SRP chapters with significantly more detail, including a statement that PSVs and PORVs are assumed to fail open if they relieve water without being qualified.

The accident at Three Mile Island, Unit 2 (TMI-2) on March 28, 1979, included failure of a PORV on the pressurizer to reclose properly during the event. Based on lessons learned from the TMI-2 accident, the NRC issued recommendations regarding performance testing of safety and relief valves used in nuclear power plants in NUREG-0578 (July 1979), "TMI-2 Lessons Learned Task Force Status Report and Short-Term Recommendations." In particular, the NRC staff recommended in Section 2.1.2, "Performance Testing for BWR and PWR Relief and Safety Valves," of NUREG-0578 that nuclear power plant licensees commit to provide performance verification by full-scale prototypical testing for all relief and safety valves.

On October 31, 1980, the NRC issued a letter to all then-operating nuclear power plants and applicants for operating licenses and holders of construction permits forwarding NUREG-0737, "Clarification of TMI Action Plan Requirements." Requirement II.D.1, "Performance Testing of

Boiling-Water Reactor and Pressurized-Water Reactor Relief and Safety Valves (NUREG-0578, Section 2.1.2)," in NUREG-0737 specified the NRC position that PWR and BWR licensees and applicants shall conduct testing to "qualify" the reactor coolant system (RCS) relief and safety valves under expected operating conditions for design-basis transients and accidents. The detailed clarification in NUREG-0737 of this NRC position specified the following:

> Licensees and applicants shall determine the expected valve operating conditions through the use of analyses of accidents and anticipated operational occurrences referenced in Regulatory Guide 1.70, Revision 2. The single failures applied to these analyses shall be chosen so that the dynamic forces on the safety and relief valves are maximized. Test-pressures shall be the highest predicted by conventional safety analysis procedures. Reactor coolant system relief and safety valve qualification shall include qualification of associated control circuitry, piping, and supports, as well as the valves themselves.
>
> A. Performance Testing of Relief and Safety Valves--The following information must be provided in report form by October 1, 1981:
>
> (1) Evidence supported by test of safety and relief valve functionability for expected operating and accident (non-ATWS) conditions must be provided to NRC. The testing should demonstrate that the valves will open and reclose under the expected flow conditions.
>
> (2) Since it is not planned to test all valves on all plants, each licensee must submit to NRC a correlation or other evidence to substantiate that the valves tested in the EPRI (Electric Power Research Institute) or other generic test program demonstrate the functionability of as-installed primary relief and safety valves. This correlation must show that the test conditions used are equivalent to expected operating and accident conditions as prescribed in the final safety analysis report (FSAR). The effect of as-built relief and safety valve discharge piping on valve operability must also be accounted for, if it is different from the generic test loop piping.
>
> (3) Test data including criteria for success and failure of valves tested must be provided for NRC staff review and evaluation. These test data should include data that would permit plant-specific evaluation of discharge piping and supports that are not directly tested.

In describing the type of review to be conducted for this regulatory position, the NRC staff stated the following:

> Pre-implementation review will be performed for EPRI and BWR test programs with respect to qualification of relief and safety valves. Also, the applicants' proposal for functional testing or qualification of PWR valves will be reviewed. Post-implementation review will also be performed of the test data and test results as applied to plant-specific situations.

In specifying the documentation required to satisfy this regulatory position, the NRC staff stated the following:

> Pre-implementation review will be based on EPRI, BWR, and applicant submittals with regard to the various test programs. These submittals should be

made on a timely basis as noted below, to allow for adequate review and to ensure that the following valve qualification dates can be met:

Final PWR (EPRI) Test Program--July 1, 1980

Final BWR Test Program--October 1, 1980

Block Valve Qualification Program--January 1, 1981

Post-implementation review will be based on the applicants' plant-specific submittals for qualification of safety relief valves and block valves. To properly evaluate these plant-specific applications, the test data and results of the various programs will also be required by the following dates:

PWR (EPRI)/BWR Generic Test Program Results--July 1, 1981

Plant-specific submittals confirming adequacy of safety and relief valves based on licensee/applicant preliminary review of generic test program results--July 1, 1981

Plant-specific reports for safety and relief valve qualification--October 1, 1981

Plant-specific submittals for piping and support evaluations--January 1, 1982

Plant-specific submittals for block valve qualification--July 1, 1982.

In a letter dated July 27, 1982, to the NRC staff, the Westinghouse Owners Group (WOG) submitted WCAP-10105 (June 1982), "Review of Pressurizer Safety Valve Performance as Observed in the EPRI Safety and Relief Valve Test Program." In WCAP-10105, the WOG indicated that the design specification for PSVs in Westinghouse-designed nuclear power plants is for steam service only. Based on a review of the EPRI test data, the WOG concluded that the valves performed with chatter, but did not identify any valve damage. (ADAMS LL Accession No. 8208190310, Microfiche 14387:191-301)

In December 1982, EPRI issued NP-2628-SR, "EPRI PWR Safety and Relief Valve Test Program – Safety and Relief Valve Test Report," that described safety and relief valve tests for types of valves in service at nuclear power plants. In particular, Section 3.5 in EPRI NP-2628-NP discusses the testing of Crosby safety valves similar to the PSVs at Braidwood and Byron, including two water tests. The report indicated chattering of the safety valves with subsequent inspection finding galled surfaces and damage to internal parts. Section 4.6 in EPRI NP-2628 discussed testing of Copes-Vulcan relief valves similar to the pressurizer PORVs at Braidwood and Byron, although the extent of water testing is not fully described. The report indicated no damage found during the inspection of the Copes-Vulcan relief valves. The report did not indicate any failures of the Crosby or Copes-Vulcan valves to reseat during the testing. (ADAMS LL Accession No. 8407130197, Microfiche 25588:082-262)

In January 1983, EPRI issued NP-2770-LD, "EPRI/C-E PWR Safety Valve Test Report," that described the testing of PWR primary system safety valves. Volume 1 provides a summary of the test program and its results. Section 4.5 of Volume 1 indicates that the following tests were performed on the Crosby 6M6 PSV: 11 steam tests with filled loop seals, 3 steam-to-water transition tests, and 2 water tests. The report states that the valve experienced chatter during the tests, and one water test had to be terminated. The individual volumes of EPRI NP-2770-LD discuss the test results for each specific PSV type. Volume 6 provides the test details for the

Crosby 6M6 PSV. (EPRI NP-2770-LD, Volume 1, was obtained as a public document from the EPRI website. EPRI NP-2770-LD, Volume 6, could not be located within ADAMS or the NRC Record Retention Files, but is available for a fee from EPRI.)

In October 1982, EPRI issued NP-2670-LD, "EPRI/Wyle Power-Operated Relief Valve Phase III Test Report," to address testing of PORVs. This document could not be located in ADAMS despite its reference by nuclear power plant licensees. See, for example, North Anna Units 1 and 2 UFSAR (Revision 51, dated September 30, 2015), Section 15.2.14, "Spurious Operation of the Safety Injection System at Power."

The NRC review of the operating license applications for Braidwood and Byron included evaluation of the TMI ~~action~~ Action Plan items as discussed in the NRC Safety Evaluation Report (SER) for Braidwood Units 1 and 2, NUREG-1002, Section 1.1, "Introduction." In this SER section, the NRC staff stated that the review and evaluation of compliance by the applicant with the licensing requirements established in NUREG-0660, "NRC Action Plan Developed as a Result of the TMI-2 Accident," and NUREG-0737 (including item II.D.1 in Table 1.1) were incorporated into the reviews summarized throughout the SER. The NRC SER for Byron Units 1 and 2, NUREG-0876, also includes discussions of the NRC staff review of the TMI ~~action~~ Action Plan items.

Appendix E, "Requirements Resulting from TMI-2 Accident," to the Braidwood/Byron UFSAR in Section E.23, "Relief and Safety Valve Test Requirements (II.D.1)," indicated that a letter dated April 1, 1982, from D. Hoffman (Consumers Power) transmitted the Safety and Relief Valve Test Report for the EPRI PWR Safety and Relief Valve Test Program. The UFSAR stated that the final evaluation of the data indicated that the relief and safety valves will perform their intended functions for all expected fluid inlet conditions. The UFSAR also indicated that the plant-specific final evaluation confirming the adequacy of the relief and safety valves had been submitted by a letter from T. Tramm, dated October 26, 1982.

In Supplement 1 to the Braidwood SER (NUREG-1002, Supplement 1, September 1986), in Section 3.9.3.3, "Design and Installation of Pressure Relief Devices," the NRC staff stated that EPRI had completed a full-scale valve testing program, and that the WOG submitted the test results in WCAP-10105 in a letter dated July 27, 1982, from O. Kinglsey to S. Chilk. (ADAMS LL Accession No. 8208190307, Microfiche 14387:189-301). The NRC staff stated that the applicant responded to a requirement to demonstrate operability of these valves through submittals dated July 1 and October 26, 1982, and December 30, 1983. On the basis of a preliminary review, the NRC staff concluded that the applicant's general approach to responding to this item was acceptable, and provided adequate assurance that the RCS overpressure protection systems at Braidwood can adequately perform their intended functions. The NRC staff stated that if the detailed review revealed modifications or adjustments to safety valves, PORVs, PORV block valves, or associated piping, were needed to ensure that all intended design margins were present, the NRC staff would require that the applicant make appropriate modifications. The NRC staff categorized this issue as a Confirmatory Item. In Supplement 5 to the Byron SER (NUREG-0876, Supplement 5, October 1984) in Section 3.9.3.3, the NRC staff provided a similar discussion of the status of the NRC review of the capability of the Byron pressurizer valves. In Supplement 8 to the Byron SER (March 1987), the NRC staff stated TMI Action Plan Item II.D.I (3.9.3.3) had been closed in Supplement 5 to the Byron SER. The NRC issued operating licenses for Byron Unit 1 in February 1985 and Unit 2 in January 1987, and Braidwood Unit 1 in July 1987 and Unit 2 in May 1988.

Following the issuance of the Byron and Braidwood operating licenses, the NRC staff provided a letter dated August 18, 1988, from L. Olshan to H. Bliss, indicating that Idaho National Engineering Laboratory (INEL) Technical Evaluation Report (TER) EGG-NTA-8028 (January 1988) provided the review of the Byron response to NUREG-0737, Item II.D.1. (ADAMS LL Accession No. 8808260355, Microfiche 46653:240-269). The NRC staff indicated that the licensee should develop and adopt plant procedures to inspect the pressurizer valves after each lift involving loop seal or water discharge. The TER described the INEL review of the EPRI testing of a PSV and PORV similar to the Byron pressurizer valves. The TER indicated that the PSV had two applicable tests: a loop seal/steam water transition test where the valve opened, chattered and stabilized to close; and a saturated water test where the valve opened with water, chattered, and stabilized. The TER indicated that the PORV opened and closed on demand in the loop seal/steam water transition test with a bending moment that was evaluated by analysis. The TER concluded that Byron provided an acceptable response to NUREG-0737, Item II.D.1. On May 21, 1990, the NRC staff provided a letter from S. Sands to T. Kovach with the Braidwood TER that included similar findings. (ADAMS LL Accession No. 9005290209, Microfiche 53927:301-330)

In January 1988, WCAP-11677, "Pressurizer Safety Relief Valve for Water Discharge During a Feedwater Line Break," provided a description of the WOG comparison of the EPRI test data with feedline break safety analyses. This report was submitted as an attachment to a response to a request for additional information (RAI) dated May 8, 1989, from the licensee of the Seabrook nuclear power plant. (ADAMS Microfiche 49775:336 – 49756:017) As discussed in the report, the WOG determined that all nuclear power plants addressed in the EPRI testing have PSVs that will operate reliably during water relief. The WOG evaluated the performance of the Crosby 6M6 PSVs during the EPRI tests, and considered that the performance involved less significant flutter (half lift motion) than the chatter (full lift motion) determined in the EPRI report. The WOG concluded that the Crosby 6M6 PSV can pass slightly subcooled water at a minimum up to three times without damage.

APPENDIX C: CONCERNS REGARDING PERFORMANCE OF PRESSURIZER VALVES UNDER WATER FLOW CONDITIONS

Westinghouse Nuclear Safety Advisory Letter

In 1993 and 1994, Westinghouse issued Nuclear Safety Advisory Letter (NSAL) 93-013 (June 30, 1993) and NSAL-93-013, Supplement 1 (October 28, 1994) to operating nuclear power plants (including Braidwood and Byron). These advisories resulted from Westinghouse's discovery that potentially nonconservative assumptions were used in the licensing analysis of the ~~Inadverent~~ Inadvertent Operation of the Emergency Core Cooling System at Power (IOECCS) event.

In NSAL-93-013, Westinghouse recommended that licensees determine if their pressurizer safety relief valves (PSRVs) are capable of closing following discharge of subcooled water. Westinghouse noted that the PSRVs might have been designed or "qualified" to relieve subcooled water. Westinghouse indicated that water relief through the power-operated relief valves (PORVs) is not a concern, because the PORV block valves can be used to isolate the PORVs if they fail to close. If the PSRVs are not designed or qualified for subcooled water relief, Westinghouse recommended that licensees re-evaluate the IOECCS event with three possible options of (1) reducing ECCS flow used in the safety analysis, (2) using a less restrictive operator response time, or (3) crediting the use of one or more PORVs to help mitigate the event.

In Supplement 1 to NSAL-93-013, Westinghouse ~~alerted~~ informed licensees ~~to~~ of a potential reduced time for operator action if a positive displacement pump is in service, and to the need to qualify the PSRVs and the piping downstream of the PSRVs and PORVs if water relief from the pressurizer is predicted.

Some licensees of operating nuclear power plants ~~alerted~~ informed the NRC ~~to~~ of their actions to address the potential concerns regarding liquid service for pressurizer safety valves (PSVs) and PORVs. A sample of actions by nuclear power plant licensees is summarized below in the "Plant-Specific Actions" section.

Additional NRC Generic Communications and Guidance

In December 2003, the NRC staff issued NRR Review Standard for Extended Power Uprates (RS-001, Rev. 0)~~.~~. Item 8 on page 7 of the review standard states that pressurizer level should not be allowed to reach a pressurizer water-solid condition.

On December 14, 2005, the NRC issued Regulatory Issue Summary (RIS) 2005~~-029~~-29, "Anticipated Transients that could Develop into More Serious Events," to notify nuclear power plant licensees of a concern identified during recent reviews of power uprate LARs. In RIS 2005~~-029~~-29, the NRC staff stated that typically Condition II event scenarios involve discharging water through relief or safety valves that are not qualified for water relief. The NRC staff stated that these valves are then assumed to fail in the open position and create a small break LOCA. The NRC staff stated that it was concerned that some licensees may be crediting PORVs without qualification for water relief and without establishing additional restrictions to ensure the availability of PORVs and block valves. The NRC staff stated that Westinghouse NSAL-93-013 allowing block valves to isolate PORVs is inconsistent with non-escalation criterion.

In proposed Revision 1 to RIS 2005-29, the NRC staff addresses the specific Condition II scenarios of chemical volume and control system (CVCS) malfunction, IOECCS event, and inadvertent opening of a PORV or PSV. Regarding the CVCS malfunction, the NRC staff states that performing only the reactivity anomaly analysis or assuming that this malfunction is not as severe as the IOECCS event is not acceptable. Regarding the IOECCS event, the NRC staff states that five of the alternative approaches in ~~NSAL-93~~NSAL-93-013 fail to meet the non-escalation criterion. The NRC staff indicated that these unacceptable alternative approaches are (1) closing the block valve, (2) assuming that the PORV is not operable, (3) addressing a stuck-open PORV or PSV as a separate Condition II event, (4) determining that a stuck-open PORV or PSV is not as severe as a small break LOCA, and (5) determining that RCS loss through PORV is made up by ECCS flow. Regarding inadvertent opening of PORV or PSV, the NRC staff states that inadvertent opening of PSV or PORV could continue as a Condition III small break LOCA and fails to meet the non-escalation criterion.

Additional General PSV/PORV Information

In August 2004, EPRI issued Report 1011047, "Probability of Safety Valve Failure-to-Reseat Following Steam and Liquid Relief - Quantitative Expert Elicitation," 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 relief through safety valves might cause increased chatter, and therefore, an increased failure rate.

In March 2011, the NRC published NUREG/CR-7037, "Industry Performance of Relief Valves at U.S. Commercial Nuclear Power Plants through 2007," based on a study by the Idaho National Laboratory. With respect to pressurizer PORVs, the report found four separate liquid relief events at four PWR plants. The report estimated 698 total demands on these PORVs during their liquid relief events with no failures to close. The report also summarized test data from EPIX for three valve types. The report indicates 2 failures of PORVs to reclose during 2070 demands, but does not specify liquid or steam service for the EPIX test information. With respect to PSVs, the report indicates 2 failures out of 4 total demands following plant scrams, but does not indicate liquid or steam service. NRC staff from the Office of Nuclear Regulatory Research provided Licensee Event Report information indicating that the 2 PSV failures involved reseating of the valves with leakage of 25 and 200 gpm, respectively. The report summarized EPIX test data for PSVs as no failures to reclose during 1805 demands.

Plant-Specific Actions

Diablo Canyon

On August 13, 1996, the licensee of the Diablo Canyon nuclear power plant submitted a report under 10 CFR 50.59 related to the potential for an IOECCS event. (ADAMS Microfiche 89419:294-322). The submittal included NSAL-93-013 and its supplement as enclosures. The licensee indicated that the PSVs had not been initially qualified for water relief, but were subsequently qualified for a brief period. The licensee indicated that WCAP-11677 was applicable and demonstrated that the PSVs were operable.

On July 2, 2004, the NRC granted a license amendment request (LAR) for Diablo Canyon that allowed credit for actuation of the PORVs in response to inadvertent safety injection (SI) actuation to avoid challenges to the PSVs. (ADAMS Accession No. ML041950300). In support of that LAR, the licensee responded on November 21, 2003, to requests for additional information (RAIs) related to the capability of the PORVs to function adequately under conditions predicted for design-basis transients and accidents (ADAMS Accession No. ML033360735). In response to an RAI regarding the design adequacy of the PORVs if the pressurizer becomes water solid, the licensee had stated that the NRC had issued a letter dated January 26, 1986, "Safety and Relief Valve Testing, NUREG-0737 Item II.D.1," that provided an SER that accepted the adequacy of the PORV and block valve design and confirmatory testing for a range of fluid conditions (full pressure steam, steam to water transition, and subcooled water fluid).

Salem

On June 4, 1997, the NRC granted a technical specification (TS) revision for the Salem nuclear power plant to ensure that the automatic capability of the PORVs to relieve pressure is maintained (ADAMS Accession No. ML011720397). In response to ~~NSAL-93~~NSAL-93-013, the licensee determined that an inadvertent ~~safety injection (~~SI~~)~~ actuation at power could cause the pressurizer to become water solid and PSVs lifting with water relief if the automatic operation of the PORVs is not made available for reactor coolant system (RCS) depressurization early in the transient. In that the Salem PSVs were not designed to relieve water, it was noted that water relief has the potential to cause the PSVs to fail in the open position.

In the course of the review of the licensee's application, the NRC staff noted that the PORVs were not designed to "safety related" standards and, thus, could not be credited for mitigation of the inadvertent SI actuation at power incident when the PORV is operating in the automatic mode. In response, the licensee proposed an upgrade of the PORVs to eliminate the possibility that a single active failure of a PORV component could prevent the mitigation of the inadvertent SI actuation at power incident. As discussed in the SER, the licensee implemented modifications to the PORV circuitry to qualify the upgraded circuitry as safety-related.

Regarding PORV performance, the licensee evaluated the PORV air accumulators for sufficient capacity for the inadvertent SI event. The licensee also reported that endurance tests had been performed with five different trims (with different trim materials) on one PORV at Wyle Laboratories to demonstrate that (1) after 2000 consecutive operations, there were no packing leaks nor packing gland adjustments required; (2) there was no diaphragm failure; and (3) the solenoid valve withstood 10,000 operations without any loss of function. Based on this information, the NRC staff concluded that the PORV performance was acceptable regarding the mitigation of an inadvertent SI event.

Millstone, Unit 3

On June 5, 1998, the NRC granted a license amendment for Millstone, Unit 3 for a TS revision to ensure that the capability of the PORVs to relieve pressure is maintained (ADAMS Accession No. ML011800207). The revised TS Bases stated that the PORVs and their associated piping have been demonstrated to be "qualified" for water relief. The PORVs prevent water relief from the PSVs for which qualification for water relief has not been demonstrated. The TS Bases also stated that the prime importance for the capability to close the block valve is to isolate a stuck-open PORV. In the SER, the NRC staff stated that the licensee notified the NRC of the issue of potential water relief through the PSVs that could lead to valve failure in LER 97-063-00 on December 31, 1997.

To provide added assurance that the PSVs will not be damaged due to water relief during an ISI event, the licensee upgraded the PORV circuitry, added additional PORV surveillance requirements, qualified the PORVs and associated piping for water relief, and made emergency procedure changes to allow plant operators additional time to terminate the event. With respect to the PORV circuitry, the NRC staff concluded that the PORV circuitry modifications qualified the PORV control circuitry as safety-related. With respect to PORV performance, the licensee reanalyzed the inadvertent SI event with the LOFTRAN computer code to demonstrate that the PORVs were qualified for water relief for approximately 1 hour. The licensee referenced EPRI testing documented in NP-2670-LD, Volume 11, that was said to generically resolve post TMI-2 issues associated with PORVs and safety valve qualification for water and steam relief, with the results from four tests of a Garrett PORV (such as used at Millstone, Unit 3) for water relief. The licensee determined that the PORVs and associated piping are qualified for 1 hour of water relief for an IOECCS event. The licensee also stated that the PORV manufacturer performed numerous cycle tests to verify the performance of the valve design, and also verified that valve seat leakage was acceptable. The licensee stated that the PORV block valves had been evaluated for water relief in accordance with the program established in response to Generic Letter (GL) 89-10, "Safety-Related Motor-Operated Valve Testing and Surveillance." The NRC staff found the licensee information regarding the qualification of the PORVs for water relief during the inadvertent SI event to be acceptable.

Callaway

On September 25, 2000, the NRC granted a license amendment for the Callaway nuclear power plant to revise the TS to change the PSV lift setting range. (ADAMS Accession No. ML003753326). To prevent water passing through the PSVs during an IOECCS event, the licensee modified and upgraded the PORV circuitry to full Class 1E to take credit for automatic action of at least one PORV during the event. These actions would prevent water relief through the PSVs. In its TS revision request dated May 25, 2000, the licensee had stated that the design function of the valves was not being changed and the conclusions documented in the NRC SER of Callaway's response to NUREG-0737 Item II.D.1 (dated September 10, 1987) are unchanged. As a result, the licensee stated that the PORVs and associated discharge piping can accommodate water relief.

Braidwood and Byron

On May 29, 1998, the Braidwood and Byron licensee proposed an amendment to its TS to take credit for the automatic operation of the PORV to provide mitigation for an IOECCS event. In the amendment request, the licensee stated that the PSVs have not been qualified to reseat after passing subcooled liquid. The licensee stated that the PORVs at Braidwood and Byron are safety-related components with safety-related actuators and accumulator tanks with PORV control circuits classified as safety-related. The licensee noted that some portions of the PORV circuitry are nonsafety-related with improvements implemented in response to GL 90-06, Resolution of Generic Issue 70, "Power-Operated Relief Valve and Block Valve Reliability" and Generic Issue 94, "Additional Low-Temperature Over Pressure Protection for Light-Water Reactors" Pursuant to 10 CFR 50.54(f)." The licensee stated that the PORV block valves are within the scope of the GL 89-10 program. In a letter dated May 13, 1999, the NRC staff provided an RAI regarding the reliance on the PORVs that documented the basis for its concerns that the PORV circuitry did not meet the single failure criterion. In response to these concerns, the licensee withdrew its TS amendment request in a letter dated July 16, 1999. No further action regarding this amendment request has been identified.

On July 5, 2000, the Braidwood and Byron licensee submitted a request for a power uprate for Braidwood and Byron to increase the maximum thermal power for each unit from 3411 megawatts thermal (MWt) to 3586.6 MWt (commonly referred as a stretch power uprate). In

RAIs, the NRC staff requested that the licensee address water solid conditions in the pressurizer because it had generally not accepted a solid pressurizer for an IOECCS event to order to avoid the potential for all three PSVs to be stuck open due to liquid relief through these safety valves. In its letter dated November 27, 2000, the licensee stated that Section 15.5.1, "Inadvertent Operation of Emergency Core Cooling System During Power Operation," of the UFSAR had been revised to credit the PSVs to pass water. The licensee discussed the EPRI testing program in response to NUREG-0737 with the results summarized in EPRI NP-2628-SR. The licensee referenced the NRC letters from L. Olshan to H. Bliss, dated August 18, 1988, and S. Sands to T. Kovach, dated May 21, 1990, transmitting the TERs with the results of the NRC's review of the Byron and Braidwood response to NUREG-0737, Item II.D.1, respectively.

On January 31, 2001, the Braidwood and Byron licensee provided a response to an RAI supplement from the NRC staff requesting the temperature of water to be passed by the pressurizer safeties and the length of time that the safeties are expected to pass water. The NRC staff also asked the licensee to discuss what EPRI tests are applicable to the Byron and Braidwood condition. In response, the licensee stated that the PSVs would close after passing water, although they may not be leaktight. The licensee stated that the leakage from up to three leaking PSVs is bounded by one fully open PSV. The licensee indicated that the EPRI testing of the Crosby safety valves in EPRI NP-2770-LD, Volumes 1 and 6, are applicable. The licensee indicated that valve chatter occurred during the tests with damage to the internals, but that the safety valve closed in response to system depressurization. The licensee stated that the Byron/Braidwood pressurizer water temperature of 590 °F is higher than the EPRI tests (530 °F). The licensee stated that the assumed length of the event is 20 minutes from initial SI signal to when the system pressure is restored below PSV lift setpoint.

In the NRC SER dated May 4, 2001, granting the Byron/Braidwood power uprate in Section 3.2, "Non-LOCA [loss-of-coolant accident] Transient Analysis," the NRC staff discussed its review of the performance of the PORVs and PSVs to discharge liquid water for approximately 20 minutes. (ADAMS Accession No. ML033040016) The NRC staff discussed the EPRI testing program with the conclusion that the safety valve closed in response to system depressurization. The NRC staff reviewed the licensee's evaluation of the performance of the PSVs for liquid water conditions. The NRC staff found that the EPRI tests adequately demonstrate the performance of the valves for the expected water temperature conditions and that there is reasonable assurance that the valves will adequately reseat following the spurious SI event. The NRC staff determined that a review of the 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. Therefore, the NRC staff found the licensee's crediting of the PSVs to discharge liquid water during the spurious SI event to be acceptable. This portion of the NRC SER was based on the specific review of PSV performance for the Byron and Braidwood power uprate request described in a memorandum dated March 15, 2001, from the NRR Reactor Systems Branch with technical input from the responsible staff member for safety valves in the NRR Division of Engineering (ADAMS Accession No. ML010740316).

As noted by the licensee, the Byron/Braidwood UFSAR at the time of the stretch power uprate (Revision 9, dated December 2002) in Chapter 15.5.1 includes PSV water relief, and references the INEL 1988 report and L. Olshan August 1988 SER. The current UFSAR Revision 15 (dated December 2014) concludes that the IOECCS event does not progress into a stuck-open PSV LOCA event. The UFSAR states that all three PSVs may lift but will reclose, and that the leakage is bounded by one fully open valve with the consequences bounded by the IOPSRV event.

On August 26, 2004, the NRC issued a license amendment for Braidwood and Byron granting an adjustment to the PSV setpoints. (ADAMS Accession No. ML042250531). In an RAI, the NRC staff requested that the licensee perform a quantitative analysis regarding PSV water cycles and relief/discharge water temperature. As for the loss of ac power (LOAC) with reactor coolant pump (RCP) seal injection event, the licensee's analysis indicated that continued injection of water into the RCS through the RCP seals would result in a water-solid pressurizer and water discharge through the PSVs. The proposed PSV setpoint tolerance assuming negative tolerance would result in a lower PSV lift setpoint. With the lower setpoint, the PSV would open earlier, and a larger number of PSV water cycles with a lower water discharge temperature could result during the transient. The licensee performed an analysis of the LOAC with RCP seal injection event, and determined the revised PSV setpoint would result in an increase of about one PSV water cycle and a reduction in the liquid discharge temperature of about 0.5 °F. A comparison of the reanalysis showed that the spurious SI event remained the limiting event since it resulted in a greater increase in the number of PSV water cycles (two cycles vs. one cycle) and a greater decrease in the PSV discharge water temperature (3.0 °F vs. 0.5 °F) than that calculated for the LOAC with RCP seal injection event. The water discharge temperature in the analysis of record for the spurious SI event was 590 °F. The lowest discharge water temperature for the spurious SI event with the revised PSV setpoint is 587 °F. The NRC staff found that the calculated water discharge temperature (587 °F) is significantly higher than the discharge water temperature of 530 °F that was used to support operability of the PSVs as discussed in the analysis of record. As a result, the NRC staff concluded that the reanalysis is acceptable to assure that the PSVs will remain operable following a spurious SI event.

On February 7, 2014, the NRC issued a license amendment for Braidwood and Byron granting a Measurement Uncertainty Recapture (MUR) power uprate. The NRC staff determined that the IOECCS event was outside of the scope of the MUR power uprate, because the licensee did not modify the Chapter 15 analyses related to PSV and PORV water relief.

Shearon Harris

On October 12, 2001, the NRC granted a license amendment to the Shearon Harris nuclear power plant for steam generator replacement and a power uprate to a maximum power level of 2900 MWt (approximately 4.5 percent). In addressing the licensee's evaluation of SRP Section 15.5.1, the NRC staff found that the analysis showed that the calculated inlet pressures and temperatures required for the PORVs and SRVs to operate in a water environment are within the valve operable ranges, and thus ensure that the PORV and SRV are operable during the transient. The valve operable ranges were previously determined by the licensee to support operability of the PORV and SRV during the discharge of subcooled water in accordance with the TMI Action Plan Item II.D.1 requirements. Based on the analysis meeting the acceptance criteria of SRP Section 15.5.1 with respect to the RCS pressure limit and departure-from-nucleate-boiling limit, the NRC staff concluded that the analysis was acceptable.

Beaver Valley

On July 19, 2006, the NRC granted an EPU to Beaver Valley Units 1 and 2 (BVPS-1 and 2) for an approximate 8-percent increase in thermal power to 2,900 MWt. In its SER (ADAMS No. ML061720376), the NRC staff stated that a specific issue which was reviewed related to the capability of the PSVs to discharge liquid and adequately reseat for a spurious SI actuation. The specific issue which the NRC staff evaluated in this regard was whether the PSVs could reasonably be expected to reseat in order to prevent the spurious SI actuation (a Condition II event) from causing a stuck-open PSV (a Condition III event). This issue was said to be further discussed in RIS 2005-29. While the PSVs are qualified to discharge steam, if the valves discharge liquid having a temperature low enough, they may not reseat properly.

Based the licensee's analysis, during a spurious SI event, the PSVs would be required to discharge steam followed by high temperature liquid after the pressurizer fills. The licensee provided plots of the pressurizer water temperatures for this event which indicated that the minimum temperature of the discharged liquid for both BVPS-1 and 2 is approximately 620 °F. To evaluate the capability of the valves to discharge and reseat, the NRC staff reviewed the available data from the full flow tests performed during the EPRI test program in 1981 for the specific PSV models representative of those installed at BVPS-1 and 2. The licensee also used the methodology contained in WCAP-11677, and determined that the minimum acceptable liquid temperature for which the PSVs are expected to successfully discharge and reseat is less than the minimum expected temperature for the spurious SI event for BVPS-1 and 2.

The NRC staff agreed that both the minimum expected liquid discharge temperature and the minimum acceptable liquid temperature had been conservatively calculated. Therefore, the NRC staff determined that, for purposes of preventing the occurrence of a more serious Condition III event, there is reasonable assurance that the PSVs would adequately discharge and reseat following a spurious SI actuation. A consideration in making this finding was that, in the unlikely event of a stuck-open PSV, the ECCS is fully capable of mitigating the resulting LOCA.

Turkey Point

On June 15, 2012, the NRC granted an EPU for Turkey Point, Units 3 and 4 that increased the thermal power level of each unit approximately 15 percent to 2644 MWt.

In the SER (ADAMS Accession No. ML11293A359), the NRC staff indicated that ECCS actuation is not a possible initiator of inadvertent increase in reactor coolant inventory because the high head SI pumps have a shut-off head below the normal RCS operating pressure. The NRC staff stated that a CVCS malfunction that increases RCS inventory was evaluated for the effects of adding water inventory to the RCS. If the pressurizer fills and causes water to be relieved through the PORVs or safety valves, then these valves could stick open and create a small break LOCA. The NRC staff stated that this would violate the acceptance criterion that prohibits the escalation of an anticipated operational occurrence (AOO) into a more serious event. Satisfaction of this acceptance criterion is demonstrated by showing that sufficient time exists for the operator to recognize the situation and end the charging flow before the pressurizer can fill. The NRC staff concluded that the licensee's analyses of IOECCS and CVCS events adequately accounted for operation of the plant at the proposed power level.

Regarding an inadvertent opening of a pressurizer relief valve, the licensee initially proposed that the consequences of this event are bounded by the small break LOCA. The NRC staff did not accept this proposed disposition. If action is not taken to secure the open valve by either closing the PORV or its block valve, the NRC staff stated that this event could escalate to a small break LOCA, which is contrary to the non-escalation criterion. When the pressurizer becomes water solid, water begins to flow through the open PORV. If the PORV is not qualified for water relief, the NRC staff stated that it is likely the PORV will not close upon demand. In this way, the NRC staff stated that the inadvertent opening of a PORV, an AOO, becomes a small break LOCA at the top of the pressurizer, a Condition III event. The NRC staff requested that the licensee address the inadvertent opening of the PORV with respect to the third criterion for a Condition II event.

The licensee provided an analysis, performed largely in accordance with NRC-approved, Westinghouse analytic methodology using the RETRAN computer code; however, this analysis was performed assuming that the PORV opened instead of the PSV. The NRC staff stated that assuming the opening of the PORV is acceptable, because the PSV is differently qualified, and reseats mechanically. An additional independent fault would be required to cause the safety valve to fail to close. The analysis indicated that the pressurizer would fill within about 240 seconds. The licensee stated that there are multiple alarms to indicate the opening of a PORV. The licensee stated that a prompt operator action is required to close the PORV and, if the PORV does not close, the operator is to close the block valve. Because the necessary actions are prompt and simple, the NRC staff agreed that there is sufficient time to secure the inadvertently open PORV without filling the pressurizer.

St. Lucie

On September 24, 2012, the NRC granted an EPU for St. Lucie, Unit 2 that increased the authorized thermal power level about 12 percent to 3020 MWt. Regarding an IOECCS event, the high pressure SI pumps are incapable during power operations of delivering flow to the RCS because the pumps' shut-off head is less than the normal RCS operating pressure of 2250 psia. Therefore, the inadvertent operation of the ECCS at power event is not a credible event and was not analyzed by the licensee for the proposed EPU. The NRC staff found that the licensee's position for not analyzing the IOECCS event to be acceptable.

Regarding a CVCS malfunction, this event increases RCS inventory as an AOO that is evaluated for the effects of adding water inventory to the RCS. The NRC staff reviewed the licensee's analyses of the CVCS malfunction event and concluded that the licensee's analyses adequately accounted for operation of the plant at the proposed power level and were performed using acceptable analytical models. The NRC staff determined that the licensee's analysis demonstrated that the pressurizer did not become water solid, assuring no water was discharged through the PSVs.

Regarding an IOPORV event, the NRC staff stated that when viewed from the mass addition perspective, this event can be evaluated in two phases: (1) an inadvertent opening of a pressurizer relief valve, followed by (2) an inadvertent ECCS actuation. In the first phase, the NRC staff stated that this event could be mitigated by closing the open pressurizer relief valve or its block valve. If the PORV or its block valve was not closed, the NRC staff stated that the IOPORV event would enter the second phase with actuation of the ECCS. Based on its review, the NRC staff determined that the pressurizer overfill analysis, available alarming system, and procedures in combination with simulator exercise result had provided reasonable assurance that the pressurizer would not be expected to fill to a water solid condition that could prevent the PORV or PSV from closing after they were open, and thus, supported that the event would not generate a more serious plant conditions, meeting the ~~the~~ non-escalation criterion. The NRC staff stated that it reviewed the licensee's analyses of the inadvertent opening of a pressurizer pressure relief valve event, and concluded that the licensee's analyses adequately accounted for operation of the plant at the proposed power level and were performed using acceptable analytical models.

The NRC staff concluded that the licensee demonstrated that the all AOO acceptance criteria are satisfactorily met.

North Anna

In UFSAR (Revision 51, dated September 30, 2015) Section 15.2.14, "Spurious Operation of the Safety Injection System at Power," the licensee for North Anna Units 1 and 2 discusses the plant response to an inadvertent SI event. In particular, UFSAR Section 15.2.14.2.3, "Event Propagation," states the following:

> Safety valve (Reference 18) and PORV (Reference 19) testing has revealed no instances of failure of the valves to reseat following water relief. Resulting leakage is within the capacity of the normal makeup system and is therefore not considered to be a small break loss of reactor coolant event. Therefore, the complete filling of the pressurizer and/or water relief via a safety valve as a result of a spurious safety injection does not constitute a failure to meet the event propagation acceptance criterion. Although primary credit for preventing the propagation of the event to a small break loss of reactor coolant event is the reseating of the PORVs and safety valves, it is noted that the PORVs (which open prior to the safety valves and, if open, preclude safety valve actuation for this event) are provided with block valves which the operator will close in the event of excessive PORV leakage.

North Anna UFSAR Section 15.2.14.3, "Conclusions," states that the complete filling of the pressurizer and/or water relief via a safety valve as a result of a spurious safety injection does not constitute a failure to meet the event propagation acceptance criterion. In UFSAR Section 15.2, "References," lists Reference 18 as EPRI NP-2770-LD, Volumes 3 and 4, "EPRI/CE PWR Safety Valve Test Reports for Dresser Safety Valve Models 31739A and 31709NA," February and March 1983; and Reference 19 as EPRI NP-2670-LD, Volume 6, "EPRI/Wyle Power-Operated Relief Valve Phase III Test Report, October 1982.

Conclusion

In conclusion, the reliance by the Braidwood/Byron licensee on the acceptable performance of the PSVs and PORVs for liquid service in response to abnormal events is not inconsistent with similar approaches by some other nuclear power plant licensees. In general, the review of activities by various nuclear power plant licensees related to PSV and PORV performance revealed reliance on EPRI, Wyle, and valve vendor testing to provide support for the performance of these valves under various service conditions. Specific certification for flow capacity of these valves for liquid service in accordance with the ASME BPV Code and National Board was not identified in the review of various justifications prepared by nuclear power plant licensees.

However, the Braidwood/Byron licensee has not addressed several potential safety and operational issues in support of its reliance on the performance of the PSVs and PORVs for the service conditions specified in the UFSAR. These issues include the following:

1. In NSAL-93-013, Westinghouse raised a potential safety concern regarding water relief through pressurizer valves. In an LAR dated May 29, 1998, proposing to upgrade the PORVs at Braidwood and Byron, the licensee stated that "the PSRVs have not been qualified to reseat after passing subcooled liquid." The licensee later withdrew this proposed LAR. However, the actions by the Braidwood/Byron licensee to address the potential safety concern raised in NSAL-93-013 to avoid water relief through PSVs (such as performed by licensees of other nuclear power plants) are not apparent.

Comment [CT]: How should these issues be discussed in the main report?

Comment [SW]: Same question. We need to be clear about why these issues do not have any bearing on our conclusions regarding backfitting. Also, we only offer conclusions about Braidwood and Byron. Would the other plants have the same or similar issues? Does this get to the possible generic nature of "the issue" as we called it earlier in our report?

2. The Braidwood/Byron UFSAR states that the performance of the pressurizer safety relief valve system and the loads on pressurizer safety relief valves, associated piping, and supports as a result of liquid discharge through the pressurizer safety relief valves, was determined to be acceptable. In support of this statement, the Braidwood/Byron UFSAR references NRC SERs dated 1988 that focused on EPRI valve testing conducted in the early 1980s in response to NUREG-0737, Item II.D.1. The licensee should discuss its current justification for determining that the pressurizer valves are capable of performing their functions consistent with the assumptions for their operating conditions described in the UFSAR. For example, the licensee should indicate the positions of the reactor system designer and applicable valve manufacturers for the performance of the pressurizer valves assumed in the UFSAR. The licensee should describe its evaluation of more recent EPRI studies that discuss the potential for failure of PSVs during liquid service based on unstable test results during the EPRI testing in the 1980s. See EPRI TR-1011047 (August 2004), "Probability of Safety Valve Failure-to-Reseat Following Steam and Liquid Relief - Quantitative Expert Elicitation," that states in Appendix B that "[b]ecause these valves are not designed for liquid flow, and because EPRI tests with subcooled liquid led to unstable conditions more often than not, the likelihood of PSV failure during an SBO [station blackout] accident would be quite high."

3. The Braidwood and Byron IST Programs specify periodic fail safe tests, exercising, and position verification testing for the PORVs; and periodic position verification testing and relief valve testing in accordance with the ASME *Code for Operation and Maintenance of Nuclear Power Plants* (OM Code), Appendix I, "Inservice Testing of Pressure Relief Devices in Light-Water Reactor Nuclear Power Plants," for the PSVs. The Braidwood and Byron IST Programs should address the IST provisions for the PSVs and PORVs consistent with the assumptions for their service conditions described in the UFSAR.

4. The Braidwood and Byron IST Programs specify exercising and position verification for the PORV block valves. In addition, the Byron IST Program specifies testing using ASME OM Code Case OMN-1, "Alternative Rules for Preservice and Inservice Testing of Active Electric Motor Operated Valve Assemblies in Light-Water Reactor Power Plants," for the PORV block valves. The licensee should verify that the PORV block valves are capable of closing to isolate the PORVs consistent with the assumptions for their service conditions described in the UFSAR.

APPENDIX D: REFERENCES

Comment [SW]: Wow! Now I understand what you meant when you kept saying you were working on the references. Presumably these, and the appendices themselves, will resolve a number of my comments requesting references.

1. **ANS 1973:** American Nuclear Society (ANS), ANS-51.1/N18.2-1973, "Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants," dated August 6, 1973. [Publicly available from ANS for a fee.]

2. **ANS 2016:** ANS, "American Nuclear Society Standards Committee Glossary of Definitions and Terminology," dated May 19, 2016. Available from ANS at http://cdn.ans.org/standards/resources/toolkit/docs/glossary-of-definitions.pdf.

3. **ComEd 1982:** Commonwealth Edison (ComEd), letter from T. R. Tramm to Harold R. Denton, U.S. Nuclear Regulatory Commission (NRC), "Byron Station Units 1 and 2; Braidwood Station Units 1 and 2; Pressurizer Safety and Relief Valves," dated October 26, 1982. [This document is not in ADAMS, but is available through the NRC Public Document Room using Accession No. 8211020633 and microfiche location 15886:171 – 15886:231.]

4. **ComEd 1998:** Commonwealth Edison Company, letter from K.L. Grasser to U.S. NRC, "Change to Credit Automatic PORV Operation for Mitigation of Inadvertent Safety Injection at Power Accident," dated May 29, 1998. [This document is not in ADAMS, but is available through the NRC Public Document Room using Accession No. 9806020180 and microfiche location A3765:275-301.]

5. **ComEd 1999:** Commonwealth Edison Company, letter from R.M. Krich to U.S. NRC, "Response to Request for Additional Information Regarding License Amendment Request to Credit Automatic Power-Operated Relief Valve (PORV) Operation for Mitigation of Inadvertent Safety Injection at Power Accident and Withdrawal of License Amendment Request," dated July 16, 1999. [This document is not in ADAMS, but is available through the NRC Public Document Room using Accession No. 9907200183 and microfiche location A8671:349-354.]

6. **ComEd 2000a:** Commonwealth Edison Company, letter from R.M. Krich to U.S. NRC, "Request for a License Amendment to Permit Uprated Power Operations at Byron and Braidwood Stations," dated July 5, 2000. ADAMS Accession No. ML003730608.

7. **ComEd 2000b:** Commonwealth Edison Company, letter from R.M. Krich 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," dated November 27, 2000. ADAMS Accession No. ML003772461.

8. **Consumers 1982:** Consumers Energy Company, letter from D. P. Hoffman to Harold R. Denton, U.S. NRC, "Safety and Relief Valve Test Report for the EPRI PWR Safety and Relief Valve Test Program," dated April 1, 1982. [This document is not in ADAMS, but is available through the NRC Public Document Room using Accession No. 8207160337 and microfiche location 13866:001 – 13869:106.]

9. **EPRI 1982a:** Electric Power Research Institute (EPRI), NP-2770-LD, "EPRI/C-E PWR Safety Valve Test Report," dated January 1982. [Publicly available from EPRI for a fee.]

10. **EPRI 1982b:** EPRI, NP-2670-LD, "EPRI/Wyle Power-Operated Relief Valve Phase III Test Report," dated October 1982. [Publicly available from EPRI for a fee.]

11. **EPRI 1982c:** EPRI, NP-2628-SR, "EPRI PWR Safety and Relief Valve Test Program – Safety and Relief Valve Test Report," dated December 1982. [This document is not in ADAMS, but is available through the NRC Public Document Room using Accession No. 8407130197 and microfiche location 25588:082-262.]

12. **EPRI 2004:** EPRI, TR-1011047, "Probability of Safety Valve Failure-to-Reseat Following Steam and Liquid Relief - Quantitative Expert Elicitation," dated August 2004. Publicly available from EPRI.

13. **Exelon 2001:** Exelon Generation Company, LLC, letter from R.M. Krich 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," dated January 31, 2001. ADAMS Accession No. ML010330145.

14. **Exelon 2002:** Exelon Generation Company, LLC, letter from R. M. Krich to U.S. NRC, "Byron/Braidwood Stations Updated Final Safety Analysis Report, Revision 9; Byron Station Technical Requirements Manual, Revision 32; Byron Station Technical Specifications Bases, Revision 31; Braidwood Station Technical Requirements Manual, Revision 26; Braidwood Station Technical Specifications Bases, Revision 36," dated December 16, 2002. [Non-publicly available; electronic files are in the NRC file center, and the transmittal letter is at ADAMS Accession No. ML023650683.]

15. **Exelon 2014:** Exelon Generation Company, LLC, "Byron/Braidwood Nuclear Stations Updated Final Safety Analysis Report (UFSAR)," Revision 15, dated December 2014. ADAMS Accession No. ML14363A393.

16. **Exelon 2015:** Exelon Generation Company, LLC, letter from J. Bradley Fewell to William M. Dean, U.S. NRC, "Appeal of Imposition of Backfit Regarding Compliance with Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.34(b), General Design Criteria (GDC) 15, GDC 21, GDC 29, and Licensing Basis," dated December 8, 2015. ADAMS Accession No. ML15342A112.

17. **Exelon 2016:** Exelon Generation Company, LLC, letter from J. Bradley Fewell to Victor M. McCree, U.S. NRC, "Appeal of Imposition of Backfit Regarding Compliance with 10 CFR § 50.34(b), General Design Criteria (GDC) 15, GDC 21, GDC 29, and Licensing Basis," dated June 2, 2016. ADAMS Accession No. ML16154A254.

18. **Northeast 1997:** Northeast Nuclear Energy Company, Licensee Event Report 97-063-00, "Millstone Nuclear Power Station Unit 3 – Inadequate Operator Response Time for Inadvertent Safety Injection (SI) Event," dated December 31, 1997. Text available through ADAMS Public Legacy Library using Accession No. 9802110033.

19. **NEI 2016:** Nuclear Energy Institute (NEI), letter from Anthony R. Pietrangelo to Victor M. McCree, U.S. NRC, "Nuclear Energy Institute Comments in Support of Exelon Generation Company Second-Level Backfit Appeal," dated June 16, 2016. ADAMS Accession No. ML16208A008.

20. **NRC 1971:** U.S. NRC, "General Design Criteria for Nuclear Power Plants," 10 CFR Part 50, Appendix A, first published February 20, 1971 (Volume 36 of the *Federal Register* (FR), page 3256), as amended.

21. **NRC 1977:** U.S. NRC, SECY-77-439, "Single Failure Criterion," dated August 17, 1977. ADAMS Accession No. ML060260236.

22. **NRC 1979:** U.S. NRC, NUREG-0585, "TMI-2 Lessons Learned Task Force Final Report," dated October 1979. ADAMS Accession No. ML061430367.

23. **NRC 1980a:** U.S. NRC, letter from Darrell G. Eisenhut, "Post-TMI Requirements," dated October 31, 1980. Text available through ADAMS Public Legacy Library using Accession No. 8012160050.

24. **NRC 1980b:** U.S. NRC, NUREG-0737, "Clarification of TMI Action Plan Requirements," dated November 1980. ADAMS Accession No. ML051400209.

25. **NRC 1981a:** U.S. NRC, NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition" [SRP], Section 15.0, "Introduction," Revision 2, dated July 1981. ADAMS Accession No. ML052350113.

26. **NRC 1981b:** U.S. NRC, NUREG-0800, Section 15.5.1 – 15.5.2, "Inadvertent Operation of ECCS and Chemical and Volume Control System Malfunction that Increases Reactor Coolant Inventory," Revision 1, dated July 1981. ADAMS Accession No. ML052350142.

27. **NRC 1981c:** U.S. NRC, NUREG-0800, Section 15.6.1, "Inadvertent Opening of a PWR Pressurizer Pressure Relief Valve or a BWR Pressure Relief Valve," Revision 1, dated July 1981. ADAMS Accession No. ML052350144.

28. **NRC 1983:** U.S. NRC, NUREG-1002, "Safety Evaluation Report Related to the Operation of Braidwood Station, Units 1 and 2," dated November 1983. ADAMS Accession No. ML12114A272.

29. **NRC 1984:** U.S. NRC, NUREG-0876, Supplement No. 5, "Safety Evaluation Report Related to the Operation of Byron Station, Units 1 and 2," dated October 1984. ADAMS Accession No. ML091340252.

30. **NRC 1985:** U.S. NRC, "Revision of Backfitting Process for Power Reactors - Parts 2 and 50," dated September 20, 1985. 50 FR 38097.

31. **NRC 1986:** U.S. NRC, NUREG-1002, Supplement No. 1, "Safety Evaluation Report Related to the Operation of Braidwood Station, Units 1 and 2," dated September 1986. [This document is not in ADAMS, but is available through the NRC Public Document Room using Accession No. 8610220157 and microfiche location 38332:281 – 38333:087.]

32. **NRC 1987:** U.S. NRC, NUREG-0876, Supplement No. 8, "Safety Evaluation Report Related to the Operation of Byron Station, Units 1 and 2," dated March 1987. [This document is not in ADAMS, but is available through the NRC Public Document Room using Accession No. 8704010167 and microfiche location 40321:134-156.]

33. **NRC 1988a:** U.S. NRC, "Revision of Backfitting Process for Power Reactors - Final Rule," dated June 6, 1988. 53 FR 20603.

34. **NRC 1988b:** U.S. NRC, letter from L. Olshan to Henry E. Bliss, Commonwealth Edison Company, "NUREG-0737, Item II.D.1, Performance Testing on Relief and Safety Valves

for Byron Station, Units 1 and 2," dated August 18, 1988. ADAMS Accession No. ML003772409 [pages 161-188 of file].

35. **NRC 1989:** U.S. NRC, Generic Letter (GL) 89-10, "Safety-Related Motor-Operated Valve Testing and Surveillance," dated June 28, 1989. ADAMS Accession No. ML031150300.

36. **NRC 1990a:** U.S. NRC, letter from S. Sands to Thomas J. Kovach, Commonwealth Edison Company, "NUREG-0737, Item II.D.1, "Performance Testing on Relief and Safety Valves for Braidwood Station, Units 1 and 2," dated May 21, 1990. ADAMS Accession No. ML003772409 [pages 189-217 of file].

37. **NRC 1990b:** U.S. NRC, GL 90-06, "Resolution of Generic Issue 70, 'Power-Operated Relief Valve and Block Valve Reliability' and Generic Issue 94, 'Additional Low-Temperature Over Pressure Protection for Light-Water Reactors,' Pursuant to 10 CFR 50.54(f)," dated June 25, 1990. ADAMS Accession No. ML031210416.

38. **NRC 1990c:** U.S. NRC, NUREG-1409, "Backfitting Guidelines," dated July 1990. ADAMS Accession No. ML032230247.

39. **NRC 1991:** U.S. NRC, letter from Anthony H. Hsia to Thomas J. Kovach, Commonwealth Edison Company, "Issuance of Amendments [for Byron and Braidwood] (TAC Nos. M77332, M77333, M77334, M77335, M77402, M77403, M77404, and M77405)," dated November 18, 1991. ADAMS Accession No. ML020860105.

40. **NRC 1997:** U.S. NRC, letter from Leonard N. Olshan, NRC, to Leon R. Eliason, Public Service Electric & Gas Company, "Salem Nuclear Generating Station, Unit Nos. 1 and 2 (TAC Nos. M97827 and M97828)," dated June 4, 1997. ADAMS Accession No. ML011720397.

41. **NRC 1998:** U.S. NRC, letter from James W. Andersen to Martin L. Bowling, Jr., Northeast Nuclear Energy Company, "Issuance of Amendment – Millstone Nuclear Power Station, Unit No. 3 (TAC No. MA1527)," dated June 5, 1998. ADAMS Accession No. ML011800207.

42. **NRC 1999:** U.S. NRC, letter from John B. Hickman to Oliver D. Kingsley, Commonwealth Edison Company, "Request for Additional Information – Byron Station, Units 1 and 2 and Braidwood Station, Units 1 and 2 (TAC Nos. MA2043, MA2044, MA2048, and MA2049," dated May 13, 1999. [This document is not in ADAMS, but is available through the NRC Public Document Room using Accession No. 9905170241 and microfiche location A8035:313-316.]

43. **NRC 2000:** U.S. NRC, letter from Jack Donohew to Garry L. Randolph, Union Electric Company, "Callaway Plant, Unit 1 – Issuance of Amendment Re: Pressurizer Safety Valves and Power Operated Relief Valves (PORVs) (TAC No. MA9080)," dated September 26, 2000. ADAMS Accession No. ML003753326.

44. **NRC 2001a:** U.S. NRC, memorandum from Frank Akstulewicz to Anthony Mendiola, "Byron and Braidwood Stations, Units 1 and 2 – Requests for a License Amendment to Permit Uprated Power Operations (TAC Nos. MA9426, MA9427, MA9428 and MA9429)," dated March 15, 2001. ADAMS Accession No. ML010740316 [non-public].

45. **NRC 2001b:** 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.

46. **NRC 2001c:** U.S. NRC, letter from N. Kalyanam to James Scarola, Carolina Power & Light Company, "Shearon Harris Nuclear Power Plant, Unit 1 – Issuance of Amendment Re: Steam Generator Replacement and Power Uprate (TAC Nos. MB0199 and MB0782)," dated October 12, 2001. ADAMS Accession No. ML012880381.

47. **NRC 2003:** U.S. NRC, Review Standard (RS) 001, "Review Standard for Extended Power Uprates," dated December 2003. ADAMS Accession No. ML033640024.

48. **NRC 2004a:** U.S. NRC, letter from Girija S. Shukla to Gregory M. Rueger, Pacific Gas and Electric Company, "Diablo Canyon Power Plant, Unit Nos. 1 and 2 – Issuance of Amendment Re: Credit for Automatic Actuation of Pressurizer Power Operated Relief Valves (TAC Nos. MB6758 and MB6759)," dated July 2, 2004. ADAMS Accession No. ML041950300.

49. **NRC 2004b:** U.S. NRC, letter from George F. Dick, Jr., to Christopher M. Crane, Exelon Generation Company, LLC, "Byron Station, Units 1 and 2, and Braidwood Station, Units 1 and 2 – Issuance of Amendments, Re: Pressurizer Safety Valve Setpoints, (TAC Nos. MB9762, MB9763, MB9760, and MB9761)," dated August 26, 2004. ADAMS Accession No. ML042250531.

50. **NRC 2005a:** U.S. NRC, SECY-05-0138, "Risk-Informed and Performance-Based Alternatives to the Single-Failure Criterion," dated August 4, 2005. ADAMS Accession No. ML051950610.

51. **NRC 2005b:** U.S. NRC, Regulatory Issue Summary 2005-29, "Anticipated Transients that Could Develop into More Serious Events," dated December 14, 2005. ADAMS Accession No. ML051890212.

52. **NRC 2004a:** U.S. NRC, letter from Timothy G. Colburn to James H. Lash, FirstEnergy Nuclear Operating Company, "Beaver Valley Power Station, Unit Nos. 1 and 2 (BVPS-1 and 2) – Issuance of Amendment Regarding the 8-Percent Extended Power Uprate (TAC Nos. MC4645 and MC4646)," dated July 19, 2006. ADAMS Accession No. ML061720274.

53. **NRC 2007a:** U.S. NRC, NUREG-0800 Section 15.0, "Introduction - Transient and Accident Analyses," Revision 3, dated March 2007. ADAMS Accession No. ML070710376.

54. **NRC 2007b:** U.S. NRC, NUREG-0800, Section 15.1.1 – 15.1.2, "Inadvertent Operation of ECCS and Chemical and Volume Control System Malfunction that Increases Reactor Coolant Inventory," Revision 2, dated March 2007. ADAMS Accession No. ML070820081.

55. **NRC 2007c:** U.S. NRC, NUREG-0800, Section 15.6.1, "Inadvertent Opening of a PWR Pressurizer Pressure Relief Valve or a BWR Pressure Relief Valve," Revision 2, dated March 2007. ADAMS Accession No. ML070820094.

56. **NRC 2011:** U.S. NRC, NUREG/CR-7037, "Industry Performance of Relief Valves at U.S. Commercial Nuclear Power Plants through 2007," dated March 2011. ADAMS Accession No. ML110980205.

57. **NRC 2012a:** U.S. NRC, letter from Jason C. Paige to Mano Nazar, Florida Power and Light Company, "Turkey Point Units 3 and 4 - Issuance of Amendments Regarding Extended Power Uprate (TAC Nos. ME4907 and ME4908)," dated June 15, 2012. ADAMS Accession No. ML11293A359.

58. **NRC 2012b:** U.S. NRC, letter from Tracy J. Orf to Mano Nazar, Florida Power and Light Company, "St. Lucie Plant, Unit 2 - Issuance of Amendment Regarding Extended Power Uprate (TAC No. ME5843)," dated September 24, 2012. ADAMS Accession No. ML12268A132.

59. **NRC 2013:** U.S. NRC, Management Directive 8.4, "Management of Facility-Specific Backfitting and Information Collection," dated October 9, 2013. ADAMS Accession No. ML12059A460.

60. **NRC 2014a:** 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.

61. **NRC 2014b:** U.S. NRC, memorandum from Samuel Miranda to Christopher P. Jackson, "Making Non-Concurrence NCP-2013-04 Public," dated February 28, 2014. ADAMS Accession No. ML14063A174.

62. **NRC 2015a:** U.S. NRC, letter from Anne T. Boland to Bryan Hanson, Exelon Generation Company, LLC, "Braidwood Station, Units 1 and 2, and Byron Station, Unit Nos. 1 and 2 – Backfit Imposition Regarding Compliance with 10 CFR § 50.34(b), GDC 15, GDC 21, GDC 29, and Licensing Basis (TAC Nos. MF3206, MF3207, MF3208, and MF3209)," dated October 9, 2015. ADAMS Accession No. ML14225A871.

63. **NRC 2015a:** U.S. NRC, Draft Revision 1 to RIS 2005-29, "Anticipated Transients that Could Develop into More Serious Events," dated July 13, 2015. ADAMS Accession No. ML15014A469. (Also published at for public comment at 80 FR 42559.)

64. **NRC 2016a:** U.S. NRC, "Backfit Review Panel Minutes – January 15, 2016 Meeting," dated January 19, 2016. ADAMS Accession No. ML16027A347 [non-public and not official record].

65. **NRC 2016b:** U.S. NRC, "Backfit Review Panel Minutes – February 5, 2016 Meeting," dated February 5, 2016. ADAMS Accession No. ML16041A465 [non-public and not official record].

66. **NRC 2016c:** U.S. NRC, "Backfit Review Panel Minutes – February 25, 2016 Meeting," dated February 25, 2016. ADAMS Accession No. ML16061A166 [non-public and not official record].

67. **NRC 2016d:** U.S. NRC, "Backfit Review Panel Minutes – March 15, 2016 Meeting," dated March 15, 2016. ADAMS Accession No. ML16096A161 [non-public and not official record].

68. **NRC 2016e:** U.S. NRC, memorandum from Anthony T. Gody, Jr., to Marissa G. Bailey, "Input for Exelon Backfit Review Panel," dated March 21, 2016. ADAMS Accession No. ML16081A405 [non-public].

69. **NRC 2016f:** U.S. NRC, memorandum from Marissa G. Bailey to William M. Dean, "Backfit Review Panel Recommendation Regarding Exelon Appeal of Backfit Affecting Braidwood and Byron Stations Regarding Compliance with 10 CFR 50.34(b), GDC 15, GDC 21, GDC 29, and the Licensing Basis," dated March 25, 2016. ADAMS Accession No. ML16082A542 [non-public].

70. **NRC 2016g:** U.S. NRC, letter from William M. Dean to J. Bradley Fewell, Exelon Generation Company, LLC, "U.S. Nuclear Regulatory Commission Response to Backfit Appeal - Braidwood Station, Units 1 and 2, and Byron Station, Units 1 and 2," dated May 3, 2016. ADAMS Accession No. ML16095A204.

71. **NRC 2016h:** U.S. NRC, ~~M~~memorandum from Victor M. McCree to Gary M. Holahan, K. Steven West, Thomas G. Scarbrough, and Michael A. Spencer, "Charter for Backfit Appeal Review Panel Associated with Byron and Braidwood Compliance with 10 CFR 50.34(b), GDC 15, GDC 21, GDC 29, and the Licensing Basis," dated June 22, 2016. ADAMS Accession No. ML16173A311 [non-public].

72. **NRC 2016i:** U.S. NRC, "An Assessment of Core Damage Frequency for Byron/Braidwood Nuclear Power Plants Supporting Backfit Appeal Review Panel," dated August 11, 2016. ADAMS Accession No. ML16214A199 [non-public].

73. **PG&E 1986:** Pacific Gas and Electric Company (PG&E), letter from H. Schierling to J.D. Shiffer, U.S. NRC, "[Diablo Canyon] Safety and Relief Valve Testing, NUREG-0737 Item II.D.1," dated January 27, 1986. [This document is not in ADAMS, but is available through the NRC Public Document Room using Accession No. 8602180450 and microfiche location 34630:271-299.]

74. **PG&E 1996:** Pacific Gas and Electric Company, letter from Gregory M. Rueger to U.S. NRC, "Forwards completed Licensing Basis Impact Evaluation of FSAR Update change which contains SE performed IAW 10CFR50.59 re reanalysis of inadvertent ECCS actuation accident," dated August 13, 1996. [This document is not in ADAMS, but is available through the NRC Public Document Room using Accession No. 9608200112 and microfiche location 89419:294-322.]

75. **PG&E 2003:** Pacific Gas and Electric Company, letter from David H. Oatley to U.S. NRC, "Response to NRC Request for Additional Information Regarding License Amendment Request 01-018, 'Credit for Automatic Actuation of Pressurizer Power Operated Relief Valves; Pressurizer Safety Valve Loop Seal Temperature,'" dated November 21, 2003. ADAMS Accession No. ML033360735

76. **Union Electric 2000:** Union Electric Company, letter from Alan C. Passwater to U.S. NRC, "Revision to Technical Specifications 3.3.2, 3.4.10, and 3.4.11 – Pressurizer Safety Valves and PORVs," dated September 25, 2000. ADAMS Accession No. ML003719636.

77. **VEPCO 2009:** Virginia Electric and Power Company (VEPCO), letter from L.N. Hartz to U.S. NRC, "North Anna Power Station Units 1 and 2 Updated Final Safety Analysis Report Revision 45," dated October 1, 2009. ADAMS Accession No. ML092810047.

78. **VEPCO 2015:** Virginia Electric and Power Company, letter from Gianna C. Clark to U.S. NRC, "North Anna Power Station Units 1 and 2 Updated Final Safety Analysis Report Revision 51," dated September 30, 2015. ADAMS Accession No. ML15296A098 [non-public].

79. **Westinghouse 1988:** Westinghouse, R.J. Dickinson and J.G. Bass, "Pressurizer Safety Relief Valve Operation for Water Discharge during a Feedwater Line Break," WCAP-11677, dated January 1988. [This document is not in ADAMS, but is available through the NRC Public Document Room using Accession No. 8905120191 and microfiche location 49755:336 – 49756:017.]

80. **Westinghouse 1993:** Westinghouse, Nuclear Safety Advisory Letter (NSAL) 93-013, "Inadvertent ECCS Actuation at Power," dated June 30, 1993. [This document is not in public ADAMS, but is available through the NRC Public Document Room using Accession No. 9608200112 and microfiche location 89419:311-315, as well as in non-public ADAMS Accession No. ML052930330, pages 2-6 of file.]

81. **Westinghouse 1994:** Westinghouse, NSAL-93-013, Supplement 1, "Inadvertent ECCS Actuation at Power," dated October 28, 1994. ADAMS Accession No. ML050320117 [pages 9-15 of file].

82. **Westinghouse 2000:** Westinghouse, NSAL-00-013, "CVCS Modeling Assumption for Loss of Offsite Power Analyses," dated August 23, 2000. ADAMS Accession No. ML103200150 [pages 131-139 of file].

83. **Westinghouse 2007:** Westinghouse, NSAL-07-10, "Loss-of-Normal Feedwater Loss-of-Offsite AC Power Analysis PORV Modeling Assumptions," dated November 7, 2007. ADAMS Accession No. ML100140163 [pages 23-26 of file]

84. **WOG 1982:** Westinghouse Owners Group (WOG), letter from Oliver D. Kingsley, Alabama Power Company, to Harold R. Denton, U.S. NRC, "NUREG-0737, Item II.D.1, 'Pressurizer Safety Valve Operability,'" dated July 27, 1982. Forwards Westinghouse WCAP-10105, "Review of Pressurizer Safety Valve Performance as Observed in the EPRI Safety and Relief Valve Test Program," dated June 1982. [This document is not in ADAMS, but is available through the NRC Public Document Room using Accession No. 8208190307 and microfiche location 14387:189-301.]

From: Spencer, Michael
To: Holahan, Gary; West, Steven; Scarbrough, Thomas; Clark, Theresa
Subject: Containment Contamination Argument
Date: Wednesday, August 17, 2016 11:21:12 AM

All,

Section 3.1.2.2 of the 2015 Backfit states:

> The licensee has not addressed the questions of how long it would take to clean up a contaminated containment, and whether the time required for completing the cleanup effort and repairing or replacing any damaged PSVs could be long enough to delay the plant's return to operation beyond the short period that is implied in the UFSAR, Chapter 15.5.1.3, definition of Condition II events.

(b)(5)

(b)(5)

Michael

Michael Spencer
Senior Attorney
U.S. Nuclear Regulatory Commission
Office of the General Counsel
Office: O15-A18
Mail Stop: O16-F3
Washington, DC 20555-0001
Phone: 301-287-9115
Fax: 301-415-3725
Michael.Spencer@nrc.gov

From:	Holahan, Gary
To:	Dean, Bill; McDermott, Brian; Evans, Michele; McGinty, Tim; Lubinski, John
Cc:	McCree, Victor; Johnson, Michael; West, Steven; Clark, Theresa; Scarbrough, Thomas; Spencer, Michael
Subject:	Exelon Backfit Appeal Panel Report
Date:	Wednesday, August 24, 2016 12:31:16 PM

NRR,

The Exelon backfit appeal panel delivered its report to the EDO and DEDO this morning (ML16236A202 and ML16236A20). The panel reviewed the NRR response to the panel's preliminary findings, but could not agree with the NRR positions. The report therefore recommends to the EDO that he support the Exelon appeal. The report will be distributed today at the EDO's request.

The EDO will make his final decision after studying the report and considering any feedback from NRR and other stakeholders.

The panel is available to discuss the report with you and respond to your questions,

Gary

From: West, Steven
To: Holahan, Gary; Clark, Theresa; Scarbrough, Thomas; Spencer, Michael
Subject: FW: Byron and Braidwood Backfit Panel Analysis
Date: Friday, August 26, 2016 8:42:03 AM

Looks like our report is circulating. Ken is the director of the Division of Reactor Safety in Region III. Ed Hackett gave me similar feedback during a CRGR meeting yesterday afternoon.

Steve

Steven West, Deputy Director
Office of Nuclear Security and Incident Response
U.S. Nuclear Regulatory Commission

301-287-3734
Steven.West@nrc.gov

From: O'Brien, Kenneth
Sent: Friday, August 26, 2016 7:58 AM
To: West, Steven <Steven.West@nrc.gov>
Subject: Byron and Braidwood Backfit Panel Analysis

Steve

I am humbled by the quality and depth of the subject item and associated report. The recommendation and report were forward to me by anther DD and of course I couldn't avoid the opportunity to read your team's effort.

The level of investigation, correlation of past and current data, and crispness of your team's line of thinking and logic is one of the best that I have ever read!

Bravo!

Ken

From:	Holahan, Gary
To:	Dean, Bill; McDermott, Brian; Evans, Michele; McGinty, Tim; Lubinski, John; McCree, Victor; Johnson, Michael; West, Steven; Clark, Theresa; Scarbrough, Thomas; Spencer, Michael
Subject:	FW: Exelon Backfit Appeal Panel Report
Date:	Wednesday, August 24, 2016 12:49:05 PM

... full ADAMS Accession numbers

Package: ML16236A198

Memo: ML16236A202; Enclosure ML16236A208

From: Holahan, Gary
Sent: Wednesday, August 24, 2016 12:31 PM
To: Dean, Bill <Bill.Dean@nrc.gov>; McDermott, Brian <Brian.McDermott@nrc.gov>; Evans, Michele <Michele.Evans@nrc.gov>; McGinty, Tim <Tim.McGinty@nrc.gov>; Lubinski, John <John.Lubinski@nrc.gov>
Cc: McCree, Victor <Victor.McCree@nrc.gov>; Johnson, Michael <Michael.Johnson@nrc.gov>; West, Steven <Steven.West@nrc.gov>; Clark, Theresa <Theresa.Clark@nrc.gov>; Scarbrough, Thomas <Thomas.Scarbrough@nrc.gov>; Spencer, Michael <Michael.Spencer@nrc.gov>
Subject: Exelon Backfit Appeal Panel Report

NRR,

The Exelon backfit appeal panel delivered its report to the EDO and DEDO this morning (ML16236A202 and ML16236A20). The panel reviewed the NRR response to the panel's preliminary findings, but could not agree with the NRR positions. The report therefore recommends to the EDO that he support the Exelon appeal. The report will be distributed today at the EDO's request.

The EDO will make his final decision after studying the report and considering any feedback from NRR and other stakeholders.

The panel is available to discuss the report with you and respond to your questions,

Gary

From: West, Steven
To: Clark, Theresa; Holahan, Gary
Cc: Spencer, Michael; Scarbrough, Thomas
Subject: My comments on Friday"s clean master
Date: Sunday, August 21, 2016 7:35:29 PM
Attachments: Backfit Appeal Panel Report (MASTER) (1) 2016 08 21 Steve.docx

Any thoughts on meeting again before Wed?

Report of the Backfit Appeal Review Panel Chartered by the Executive Director for Operations to Evaluate the June 2016 Exelon Backfit Appeal

August XX, 2016

ADAMS Accession No. MLXXXXXXXXX

TABLE OF CONTENTS

1 BACKGROUND

On June 22, 2016,[1] in accordance with NRC Management Directive (MD) 8.4,[2] the NRC Executive Director for Operations (EDO) established a Backfit Appeal Review Panel (Panel) to review the appeal by Exelon Generation Company, LLC (Exelon or the licensee) of the U.S. Nuclear Regulatory Commission (NRC) staff's determination that a backfit is necessary at Braidwood Station, Units 1 and 2 (Braidwood) and Byron Station, Units 1 and 2 (Byron), as well as the NRC staff's application of the compliance backfit exception provided in Title 10 of the *Code of Federal Regulations* (10 CFR), Section 50.109, "Backfitting."

This backfit determination is documented in an October 9, 2015, letter (referred to as the Backfit Letter).[3] The letter describes the NRC staff's review of licensing basis documents for Byron and Braidwood. The NRC staff determined that Byron and Braidwood were not in compliance with the plant-specific design bases and ~~several~~ the following NRC regulations:

- General Design Criterion (GDC) 15, "Reactor coolant system design," in 10 CFR 50, Appendix A, "General Design Criteria for Nuclear Power Plants"
- GDC 21, "Protection system reliability and testability"
- GDC 29, "Protection against anticipated operational occurrences"
- Paragraph (b) of 10 CFR 50.34, "Contents of applications; technical information"

Specifically, the NRC staff determined that Byron and Braidwood do not comply with provisions in American Nuclear Society (ANS) Standard 51.1/N18.2-1973[4] for ensuring that ANS Condition II events[5] do not progress to more serious ANS Condition III events following water discharge[6] through certain valves. The NRC staff acknowledged that the NRC staff position differed from a previous staff position documented in a May 4, 2001, safety evaluation (SE) supporting a stretch power uprate (referred to as the Uprate SE).[7] However, the NRC staff determined that the backfitting was justified under the compliance exception in 10 CFR 50.109(a)(4)(i). The NRR staff directed the licensee ~~was directed~~ to take action to resolve the non-compliance.

On December 8, 2015, the licensee appealed the NRC staff's decision to the Director of the Office of Nuclear Reactor Regulation (NRR), stating its disagreement with the NRC's conclusion that the compliance exception to the backfit rule appliesd in this case, ~~and~~ while noting that the NRC staff hasd twice approved the underlying analysis.[8] The ~~referenced~~ approvals referenced

Comment [SW]: Okay, this is too picky to be treated as a serious comment, but it is interesting. In addition to this first use, there is only one other instance in this report where we put Braidwood before Byron. (It's within quoted material at the bottom of page 8). There are 88 instances of "Byron and Braidwood," including this one.

Comment [SW]: I not sure if this edit is the proper fix for the citation, but we should make it clear that we're referring to Part 50.

[1] NRC 2016e (Author and year citations in footnotes refer to the designation of references in Appendix D to this report.)
[2] NRC 2013
[3] NRC 2015b – referred to as the Backfit Letter in the remainder of the report
[4] ANS 1973
[5] Specifically, inadvertent operation of the emergency core cooling system (IOECCS), malfunction of the chemical and volume control system (CVCS), and inadvertent opening of a pressurizer safety or relief valve.
[6] For consistency in this report, the Panel usesd the phrase "water discharge" rather than "water relief" or "liquid discharge" (except in direct quotes), as this is the phrase used in the Westinghouse documents that raised the issue addressed in this report.
[7] NRC 2001b – referred to as the Uprate SE in the remainder of the report
[8] Exelon 2015 – referred to as the NRR Appeal in the remainder of the report

by the licensee were an August 26, 2004, license amendment associated with pressurizer safety valve (PSV) setpoints[9] and the above-referenced Uprate SE. In a letter dated May 3, 2016, the NRC responded to the licensee's appeal and reaffirmed its decision that the backfit per the compliance exception provisions of 10 CFR 50.109(a)(4)(i) is appropriate.[10]

On June 2, 2016, the licensee again appealed the NRC staff's decision, this time to the EDO.[11] The purpose of this report by the Backfit Appeal Review Panel is to provide information and recommendations to support the EDO's decision ~~of~~ on the ~~EDO~~ appeal.

Comment [SW]: Did we include the UFSAR as a reference and have we been clear about which edition we've reviewed?

Comment [SW]: I know this is defined in footnote 5, above, but should we also spell out the first use of acronyms in the body of the report?

And, I hate to ask, should we include a list of acronyms? We've used quite a lot of them.

1.1 Conduct of the Panel's Review

In order to establish a technically sound, well informed, and legally defensible basis for its recommendations, the Backfit Appeal Review Panel undertook a review of the relevant documents in this case. This included the licensee and NRC staff letters mentioned above; the Uprate SE and the Setpoint SE; and a June 16, 2016, letter from the Nuclear Energy Institute (NEI)[12] supporting the Exelon backfit appeal to the EDO. The Panel also reviewed many other related documents, which fall into five broad categories:

- The Backfit Rule (10 CFR 50.109), related court actions, and Commission and staff guidance on application of the Backfit Rule
- Docketed communications for Byron and Braidwood from 1982 to the present, including license amendment requests (LARs) by the licensee, NRC-issued license amendments, NRC requests for additional information (RAIs), licensee responses, meeting summaries, NRC SEs, and the licensee's Updated Final Safety Analysis Report (UFSAR)
- NRC guidance relevant to the analysis of IOECCS events over the period of 1981 to the present, including Standard Review Plan (SRP) Section 15.0, Sections 15.5.1 – 15.5.2, and Section 15.6.1[13]
- Westinghouse Nuclear Safety Advisory Letter (NSAL) 93-013[14] and its Supplement 1[15], as well as docketed communications regarding actions taken by other licensees in response to Westinghouse NSAL-93-013
- The history of NRC and industry activities related to power operated relief valves (PORVs), their block valves, and PSVs (including Three Mile Island (TMI) Action Plan Items II.D.1, II.D.3, II.G.1, and II.K.3 as documented in NUREG-0737[16], as well as Generic Letter 89-10[17] and its supplements), Electric Power Research Institute (EPRI) valve testing, and operating experience (NUREG/CR-7037[18])

[9] NRC 2004b – referred to as the Setpoint SE in the remainder of the report
[10] NRC 2016d – referred to as NRR Appeal Decision in the remainder of the report
[11] Exelon 2016 – referred to as EDO Appeal in the remainder of the report
[12] NEI 2016
[13] NRC 1981a, NRC 1981b, NRC 1981c, NRC 2007a, NRC 2007b, and NRC 2007c
[14] Westinghouse 1993
[15] Westinghouse 1994
[16] NRC 1980b – referred to as the TMI Action Plan in the remainder of the report~~;~~. ~~l~~Lessons learned from TMI were also presented in NUREG-0585 (NRC 1979)
[17] NRC 1989
[18] NRC 2011

In addition to the document review, the Panel had the benefit of meetings with NRR (both the Division of Safety Systems and the Division of Engineering), the Office of the General Counsel, and the NRC Committee to Review Generic Requirements (CRGR). Both Exelon (Bradley Fewell, Senior Vice President of Regulatory Affairs) and NEI (Tony Pietrangelo, Senior Vice President and Chief Nuclear Officer) declined offers for a public meeting, but indicated a willingness to provide information if the Panel identified the need. The Panel did not identify a need for additional information from either Exelon or NEI to complete its review, which is ~~summarized below and~~ documented ~~in the attached report~~ herein.

At the request of the Panel, the Office of Nuclear Regulatory Research (RES) conducted risk analyses using the NRC's Standardized Plant Analysis Risk model for Byron Unit 1.[19] These analyses informed the Panel's response to the question from the EDO regarding the risk significance of the relevant accident sequences.

1.2 Proposed Compliance Backfit and Exelon Appeals

In the Backfit Letter, the NRC staff informed Exelon that it had determined that Byron and Braidwood are not in compliance with GDCs 15, 21, and 29; 10 CFR 50.34(b); and the plant-specific design bases that were expected to demonstrate there will be no progression of ~~Category~~ ANS Condiition II events to more serious ~~Category~~ ANS Condition III or IV events. The NRC staff stated that based on its review of Byron and Braidwood UFSAR Sections 15.5.1, 15.5.2, and 15.6.1, the UFSAR predicts water discharge through a valve that is not "qualified" for water discharge. Therefore, the NRC staff concluded that the UFSAR does not contain analyses that demonstrate that the plants' structures, systems, and components (SSCs) ~~will~~ meet the design criteria for ANS Condition II ~~faults~~ events as stated in Byron and Braidwood UFSAR Section 15.0.1.2. Based on the SE attached to its letter,[20] the NRC staff found that the licensee must take action to resolve the non-compliance.

The Backfit SE addressed three accident analyses in Chapter 15 of the Byron and Braidwood UFSAR: (1) IOECCS; (2) CVCS malfunction that increases reactor coolant inventory; and (3) inadvertent opening of a pressurizer safety or relief valve (IOPORV). The NRC staff noted that each ANS Condition II event must be shown to meet the following:

Comment [SW]: This is the first use in the body of the report. (Also defined in footnote 5, above.)

1. no fuel damage,
2. no overpressure of the reactor coolant system (RCS) or main steam system, and
3. no progression into an event of a more serious category without another independent fault.

Regarding an IOECCS, the NRC staff stated in Section 3.1.2.1 of the Backfit SE that use of the block valve to isolate a stuck-open PORV was unacceptable. The NRC staff stated that Westinghouse had recommended this approach in 1993, and that the NRC staff rejected this approach in 2005 (RIS 2005-29[21]).

In Section 3.1.2.4 of the Backfit SE, the NRC staff stated that the Byron and Braidwood IOECCS analysis depend~~s~~ed on water discharge through the PSVs. The NRC staff faulted the

[19] NRC 2016f
[20] Referred to as the Backfit SE in the remainder of the report.
[21] NRC 2005b

licensee for "not appl[ying] the single-failure assumption" and stated that the following information was necessary to support water qualification of the PSVs:

1. In accordance with the American Society of Mechanical Engineers (ASME) *Boiler and Pressure Vessel Code* (BPV Code), Section III, provide the original Overpressure Protection Report defining operating conditions and required relief capacities, and manufacturer's certification and test results

2. In accordance with the ASME *Code for Operation and Maintenance of Nuclear Power Plants* (OM Code), provide inservice test history for PSVs, including water and steam tests, or provide correlation test for alternative test fluid.

Regarding a CVCS malfunction, the NRC staff stated in Section 3.2 of the Backfit SE that the licensee had not provided an analysis for the CVCS malfunction that increases reactor coolant inventory that demonstrate~~s~~d the plants' ability to meet the requirements of an ANS Condition II event.

Regarding an IOPORV, the NRC staff stated in Section 3.3 of the Backfit SE that the licensee had not provided an analysis for the IOPORV that extends long enough into the transient to demonstrate the event would not transition from an ANS Condition II event to an ANS Condition III event.

In the Backfit SE, the NRC staff referenced Millstone[22] and Callaway[23] license amendments as examples of licensees upgrading PORVs for water discharge; a Beaver Valley extended power uprate (EPU) license amendment[24] as an example of qualifying PORVs for water discharge; and Turkey Point[25] and St. Lucie Unit 2[26] EPU amendments as additional precedent in support of the backfit decision.

In the NRR Appeal, Exelon asserted that the NRC had not justified invoking the compliance exception to the backfit rule. Exelon stated that the NRC had approved its IOECCS analysis in both the Uprate SE and the Setpoint SE.

In the NRR Appeal Decision, the NRC staff stated that the previous staff approvals in 2001 and 2004 were inconsistent with the Agency's general position on the known and established standard at issue~~.~~; in this case the non-progression of ANS Condition II events to higher level events. The NRC staff stated that the fact that the NRC staff were aware of references to EPRI reports on the ability of these non-water qualified PSVs to reseat in certain circumstances ~~is~~ was not sufficient to support the licensee's position on the compliance backfit.

In the EDO Appeal, Exelon stated that the NRC had misidentified the "known and established standard" at issue as the prohibition of ANS Condition II events progressing to ANS Condition III events. Exelon asserted that the standard in question concerns what is necessary to "qualify" valves for water discharge. Exelon contended that this standard is the EPRI testing and analysis, and that the NRC ~~has~~ had agreed that Byron and Braidwood meet this standard. Exelon also contended that the change in NRC staff position on prior approvals ~~is~~ was not a

[22] NRC 1998
[23] NRC 2000
[24] NRC 2006
[25] NRC 2012a
[26] NRC 2012b

mistake of fact, but rather a new or modified interpretation of compliance with NRC requirements, for which use of the compliance exception provided for in the Backfit Rule ~~is~~ was not appropriate.

1.3 Backfit Rule and the Compliance Exception

Backfitting is defined by 10 CFR 50.109(a) as:

> … the modification of or addition to systems, structures, components, or design of a facility; or the design approval or manufacturing license for a facility; or the procedures or organization required to design, construct or operate a facility; any of which may result from a new or amended provision in the Commission's regulations or the imposition of a regulatory staff position interpreting the Commission's regulations that is either new or different from a previously applicable staff position …~~.~~

Unless one of three specified exceptions apply, the NRC may impose a backfit only if it performs a backfit analysis in accordance with 10 CFR 50.109(a)(2) and determines in accordance with 10 CFR 50.109(a)(3) "that there is a substantial increase in the overall protection of the public health and safety or the common defense and security to be derived from the backfit and that the direct and indirect costs of implementation for that facility are justified in view of this increased protection."

Section 50.109(a)(4) sets forth the three exceptions to the requirements of 10 CFR 50.109(a)(2) and (a)(3). The first exception, the compliance exception, applies if the "modification is necessary to bring a facility into compliance with a license or the rules or orders of the Commission, or into conformance with written commitments by the licensee." The second and third exceptions relate to actions necessary to ensure adequate protection or to actions that involve defining or redefining adequate protection.

The Commission explained its intended application of the compliance exception in the Statements of Consideration (SOC) accompanying the 1985 final rule amending 10 CFR 50.109:[27]

> The compliance exception is intended to address situations in which the licensee has failed to meet known and established standards of the Commission because of omission or mistake of fact. It should be noted that new or modified interpretations of what constitutes compliance would not fall within the exception and would require a backfit analysis and application of the standard.

In the same SOC, the Commission acknowledged that staff interpretations of rules are not legally binding, but the Commission also stated that "staff interpretations of broadly stated rules are often necessary to give a rule effect and in some instances may be a causal factor in initiating a backfit."[28]

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[27] NRC 1985, at 38103

[28] NRC 1985, at 38102. The 1985 backfit rule was vacated by a Federal court on grounds unrelated to the compliance backfit exception. See *Union of Concerned Scientists v. U.S. Nuclear Regulatory Com'n*, 824 F.2d 108, 119-20 (1987). In 1988, the Commission amended the backfit rule (NRC 1988b) to address the court's concerns, but did not change the 1985 rule's compliance exception provision. Thus, the quoted statements from the 1985 rule are the applicable expression of Commission intent regarding compliance backfits.

By its terms, the compliance exception applies to actions necessary for compliance with rules, licenses, and orders, or for conformance with written commitments.[29] Also, the Commission explicitly acknowledged the importance of staff interpretations of rules in the regulatory process. Thus, the Panel understands the term "known and established standard" to include standards established in rules, licenses, orders, and written commitments, and NRC interpretations of rules. Some standards may be broad-based, while others may apply only to a limited number of plants. As stated in NUREG-1409, "[i]nformal or formal communications to one licensee are not official positions to all licensees. ... Orders, licenses, and written commitments are applicable only to a particular licensee."

The failure to meet a known and established standard is grounds for a compliance backfit if this failure is due to "omission or mistake of fact." Thus, if a licensee obtains NRC approval of an alternative to a specific standard set forth in guidance, that standard and guidance could not be used to support a compliance backfit unless the NRC's approval of the alternative was based on an omission or mistake of fact. "Known and established standards" are to be distinguished from "new or modified interpretations of what constitutes compliance," which do not fall within the compliance exception. The Panel understands the term "new or modified interpretations" to include situations where the NRC staff has, in effect, "changed its mind" on how to interpret the language of a requirement or on how much assurance is necessary to conclude that the requirement is met. Levels of assurance might be established in terms such as acceptable probabilities or consequences, conservative assumptions, or sufficient margin.

Additional background information on the Backfit Rule and the compliance exception is provided in Appendix A to this report.

1.4 A Brief History of Pressurizer Valve Issues

Appendix B to this report provides a summary of the NRC and industry's testing, evaluation, and other consideration of PORVs and PSVs since the TMI Unit 2 (TMI-2) accident in 1979. This historical review provides context for discussion of valve "qualification" in the Backfit SE. It also provides the basis for the Panel's conclusions regarding the "known and established standard" for "qualification" in the context of ~~the~~ TMI Action Plan ~~item~~ Item II.D.1 and subsequent activities, as well as how it should be interpreted in the Byron and Braidwood licensing basis.

In light of the NRC staff's assertion that the licensee had not applied the "single-failure assumption" as noted above, the Panel also considered the applicability of the single failure criterion to PSVs. The Panel expended considerable effort in searching for an answer to what appears to be a simple question: "Are PSVs active components subject to the single failure criterion, or are they passive components exempt from ~~it~~ the single failure criterion?" NRR staff have taken the position that PSVs have consistently been treated as active components.

In the Panel's evaluation of the treatment of PSV failure potential (Section 3 below), an historical perspective is provided. In general, the Panel found that the classification of a component as "active" or "passive" depends on its design, application, and function. For example, passive components almost always do not need external power; usually do not need an external actuator (e.g., signal)[30]; sometimes do not involve any mechanical motion (e.g., movement of a

[29] NUREG-1409 (NRC 1990c) defines written commitments broadly to include the "final safety analysis report, licensee event reports, and docketed correspondence, including responses to NRC bulletins, generic letters, inspection reports, or notices of violation and confirmatory action letters."
[30] For example, SECY-77-439 (NRC 1977) states: "Examples [of passive failures in fluid systems] include

valve disc)[31]; and sometimes do not involve any motion, either fluid or mechanical (e.g., piping). International Atomic Energy Agency (IAEA) TECDOC-1624 states that "[s]afety related terms such as passive and inherent safety have been widely used, particularly with respect to advanced nuclear plants, generally without definition and sometimes with definitions inconsistent with each other." This guidance further defines four level of "passivity" to ~~"to~~ help eliminate confusion and misuse of the terms by members of the nuclear community." In addition, SECY-05-0138[32] also acknowledge~~s~~d and discusse~~s~~d inconsistencies in the use and application of the term "passive." (See Section 3.10, below.)

The introduction to the GDCs and the related footnote defined the applicability of the single failure criterion in terms of electrical versus fluid systems, and active versus passive components. Neither the GDCs nor NRC guidance define which characteristics of passive components are necessary to make a component exempt from the single failure criterion. Some examples are clear: pipes are passive components and pumps and motor-operated valves that operate to perform their safety functions are active components. As discussed in Section 3.6 below, check valves might be classified as active or passive components depending on certain specific considerations.

With respect to PSVs, the ASME BPV Code applicable to Byron and Braidwood includes requirements for overpressure protection that relate to the single failure criterion through several specific design and construction requirements. As a result, the PSVs are conservatively sized with sufficient margin to accommodate a single failure although the single failure criterion is almost never explicitly discussed or applied in accident analyses. The Byron and Braidwood UFSAR states that "adequate overpressurization protection is provided by the three installed safety valves." Neither the UFSAR system descriptions nor the safety analyses provide detailed discussions of potential PSV failures or their consequences. The principal discussion of potential PSV failures in the accident analyses occurs in the evaluation of an inadvertent opening of a PSV in UFSAR Section 15.6.1.

Most relevant for the current issue, the Byron and Braidwood UFSAR analyses of overpressure events (e.g., loss of load, loss of feedwater) do not apply the single failure criterion to cause a PSV to stick open (i.e., fail to reseat) when opening on steam flow. In addition, the UFSAR Feedwater System Pipe Break analysis (Chapter 15.2.8) does not apply the single failure criterion to cause a PSV to stick open either during steam discharge or during water discharge. A survey of other Westinghouse-designed plants showed that this treatment of PSV valve performance during anticipated operational occurrences (AOOs, similar to ANS Condition II events) and postulated accidents (similar to ANS Condition IV events) has been consistent and without any identified exceptions.[33]

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the failure of a simple check valve to move to its correct position when required, the leakage of fluid from failed components, such as pipes and valves—particularly through a failed seal at a valve or pump—or line blockage. Motor-operated valves which have the source of power locked out are allowed to be treated as passive components."

[31] For example, NUREG-1800 (NRC 2001c) states that "'[p]assive' structures and components, for the purpose of the license renewal rule, are those that perform an intended function … without moving parts or without a change in configuration or properties … 'passive' may also be interpreted to include structures and components that do not display 'a change of state.'"

[32] NRC 2005a

[33] Examples include Watts Bar (NRC 1982 and TVA 1983), North Anna (NRC 1976), and AP1000 (Westinghouse 2011).

Comment [SW]: Should we include this as a reference with title and date? Also, I don't think this document has any regulatory standing so we're referencing it and highlighting one of its observations to further illustrate our point that smart people working with the same set of facts can be inconsistent and arrive at different answers?

Comment [SW]: Too many to's?

1.5 History and Review of Westinghouse NSAL and Related Activities

Appendix C to this report provides the Panel's review of the issues identified by Westinghouse in NSAL-93-13 and its Supplement 1, how various licensees have responded to these issues, and how the NRC was involved in reviewing and approving these actions. This review provides the basis for the Panel's conclusions related to the approach taken by Byron and Braidwood to address these issues in their licensing basis, as well as on the "known and established standard" for event escalation from ANS Condition II to ANS Condition III, referred to hereafter as the "non-escalation position."

Comment [SW]: According to word search, except for the first paragraph, this is the only case in the report where we put Braidwood before Byron – and that's because it's part of a quote.

2 SUMMARY OF THE APPEAL REVIEW PANEL FINDINGS

For the reasons provided in Section 3, the Panel ~~concludes~~ concluded that in 2001 and 2004 and at present, the known and established standard of the Commission is 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. The Panel also ~~concludes~~ concluded that, in preparing the Uprate SE and the Setpoint SE, the NRC staff exercised reasonable and well-informed engineering judgment when the NRC staff concluded that the PSVs were unlikely to stick open. The non-escalation position does not establish specific standards for valve qualification, so the non-escalation position, standing alone, provides no basis for rejecting the licensee's reliance on EPRI valve testing. Moreover, the Panel ~~finds~~ found that no mistake or error occurred in the licensee's or previous staff's reliance on the EPRI testing program that included an evaluation of water discharge through pressurizer valves.[34] Therefore, the Panel also ~~concludes~~ concluded that the staff's position on valve qualification in the Backfit SE is a new or modified interpretation of what constitutes compliance.

The panel also concluded that the issue of pressurizer valve performance following water discharge appears to have generic applicability, and is not specific to only Byron and Braidwood. The Panel believes that resolution of this issue would have benefited from consideration of the generic nature of the issue through the appropriate NRC processes. The Panel included additional information about this finding in Section 6 and Appendices B and C.

3 DISCUSSION

The compliance exception to the Backfit Rule is intended to address failures to meet known and established Commission standards because of omission or mistake of fact. New or modified interpretations of what constitutes compliance do not fall within the exception. The Panel reviewed and evaluated the information referenced in this report to determine if, in 2001 and 2004, there was a known and established standard of the Commission relating to the potential for PSVs to fail following water discharge during IOECCS events.

In addition, the Panel considered the issue of "known and established standards of the Commission" as it relates to "event escalation." In the NRR Appeal Decision, the ~~NRC staff~~ NRR appeal panel stated that the Backfit SE "showed that the approvals at issue for Braidwood and Byron were inconsistent with the Agency's general position on the known and established standard at issue, in this case the progression of [ANS] Condition II events." The Panel recognizes that the non-escalation position, although not included in NRC regulations, is widely

[34] "Pressurizer valves" is used in this report to refer to either PORVs or PSVs when discussing issues common to both types of valves.

referenced in reactor licensing bases as an approach for addressing AOOs and postulated accidents as articulated in the GDCs. The non-escalation position is incorporated in Section 15.0.1.2 of the Byron and Braidwood UFSAR as "By definition, these faults (or events) do not propagate to cause a more serious fault, i.e., [ANS] Condition III or IV events."

~~Neither~~ Exelon ~~nor~~ and the Panel ~~disputes~~ agree that the non-escalation position is now, and was in 2001 and 2004, a part of the licensing basis of both Byron and Braidwood. ~~The~~ In addition, the Panel supports the NRC staff's view that non-escalation (from ANS Condition II to ANS Condition III or IV) is a known and established standard applicable to Byron and Braidwood. However, the Panel also agrees with Exelon that the fundamental issue is not the non-escalation position, as the NRR staff contends, but rather the appropriate standard for PSV water discharge. In the absence of a PSV failure to reseat, the concerns articulated by the staff in the backfit related to event classification, event escalation, and compliance with 10 CFR 50.34(b) and GDCs 15, 21, and 29 would no longer be at issue.

The Panel's evaluation of the treatment of PSV failure potential includes an assessment of multiple relevant references, which are discussed chronologically in the sections that follow.

3.1 General Design Criteria (1971)

In 1971, the Atomic Energy Commission published the GDCs, which had been under development since 1965.[35] The introduction to 10 CFR Part 50, Appendix A, addresses "Single Failure" in the section on Definitions and Explanations. The paragraph on single failures includes a footnote stating: "The conditions under which a single failure of a passive component in a fluid system should be considered in designing the system against a single failure are under development" (emphasis added).

3.2 Commission Paper on Single Failure (1977)

In response to several staff concerns and differing views on the subject of application of the single failure criterion, the Acting Director of NRR issued SECY-77-439 "[t]o inform the Commission of the present status and future use of the Single Failure Criterion as a tool in the reactor safety process."[36] In part, that paper addressed the application of the single failure criterion to passive components in fluid systems, stating that "[a]pplication of the [single failure] concept is complicated by the interrelationships between the various fluid and electrical systems and their supporting auxiliaries in a nuclear power plant. Furthermore, there is a need to stipulate the events and associated assumptions which must be considered during application of the Single Failure Criterion."

SECY-77-439 specifically spoke to how "additional passive failures"—that is, failures in addition to the initiating event—had been and should be addressed, stating (with emphases added):

> During subsequent years [since the single failure footnote quoted above was published] staff assumptions regarding the nature of passive failures which should be considered have not been completely consistent and there has been some disagreement. However, on the basis of the licensing review experience accumulated in the period since 1969, it has been judged in most instances that the probability of most types of passive failures in fluid systems is sufficiently

[35] AEC 1971
[36] NRC 1977

small that they need not be assumed in addition to the initiating failure in application of the Single Failure Criterion to assure safety of a nuclear power plant.

Furthermore, SECY-77-439 provides definitions and examples for distinguishing between active and passive failures. Among these examples, SECY-77-439 cites "the failure of a simple check valve to move to its correct position when required" as a passive failure. Of the examples cited in SECY-77-439, the check valve example is most similar from a mechanical perspective to the PSV failure addressed in the Backfit SE, as explained below in the discussion of SECY-94-084.

SECY-77-439 also stresses the use of engineering judgment relating to the probability of component failure and does not suggest that valve "certification" or "qualification" in accordance with ASME standards should be invoked as the basis for such decisions.

3.3 TMI Action Plan Item II.D.1 (1980)

As an element of the TMI Action Plan, the NRC staff required licensees to address the capability of relief and safety valves to perform their intended functions without failure. Specifically, Item II.D.1 states that "[p]ressurized-water reactor [PWR] and boiling-water reactor [BWR] licensees and applicants shall conduct testing to qualify the [RCS] relief and safety valves under expected operating conditions for design-basis transients and accidents." With reference to planned EPRI testing and other generic industry test programs, NUREG-0737 specified provisions for then-operating nuclear power plants and applicants for operating licenses and holders of construction permits to address the TMI Action Plan items, including Item II.D.1. NUREG-0737 stated, for the performance testing of relief and safety valves for Item II.D.1, that "[t]he testing should demonstrate that the valves will open and reclose under the expected flow conditions."

Although limited in scope, the EPRI test results did not identify any generic issues with PSVs or PORVs sticking open following water discharge. The NRC staff approvals summarized below show that the word "qualify" in this TMI Action Plan item was not intended to refer to ASME valve certification or qualification. Instead, "qualify" was used in a less formal sense to refer to a reasonable judgment that the valve would open to relieve pressure and then reliably reseat. As referenced in NUREG-0737, the EPRI test program was the widely used approach to address TMI Action Plan Item II.D.1 at PWR nuclear power plants. The Westinghouse Owners Group submitted WCAP-10105 to the NRC in 1982 to demonstrate the acceptability of the EPRI testing program for PSVs and PORVs in Westinghouse-designed PWRs.[37]

3.4 NRC Closure of TMI Action Plan Item II.D.1 for Byron and Braidwood (1988-1990)

A 1988 letter from the NRC staff to the licensee for Byron found the licensee's reliance on EPRI testing of PSVs to be acceptable.[38] The 1988 SE states that the test program was designed "[t]o reconfirm the integrity of the overpressure protection system and thereby assure that the [GDCs] are met." As discussed in Appendix B to this report, the 1988 SE described ~~d~~ the staff's evaluation of the PSVs and PORVs for feedwater line break accidents that would include water discharge, and determined that the EPRI tests were applicable to the Byron and Braidwood

[37] WOG 1982

[38] NRC 1988c, referred to as the 1988 SE

PSVs and PORVs. Based on the NRC staff and contractor review, the 1988 SE found that the performance of the PSVs and PORVs was acceptable based on the EPRI tests.

For the specific extended high pressure injection event, the 1988 SE states that water discharge through the PSVs and PORVs could be disregarded because of the long time available for operator action. However, the SE addressed water discharge through the PSVs and PORVs as part of the feedwater line break evaluation.

In the cover letter for the 1988 SE, the NRC staff states that the licensee should develop and adopt plant procedures to inspect the pressurizer valves after each lift involving loop seal or water discharge. The 1988 SE contains no reference to or suggestion of a need for certification of these valves in accordance with the ASME BPV Code for water discharge capability. In 1990, the NRC staff also found the use of the EPRI test program ~~was also found~~ similarly acceptable for Braidwood.[39]

3.5 Westinghouse NSAL-93-013 and Supplement 1 (1993-1994)

In 1993, Westinghouse sent NSAL-93-013 to operating nuclear power plants in response to its discovery that potentially non-conservative assumptions had been used in the licensing analysis of the IOECCS event. Westinghouse recommended that licensees determine if their pressurizer safety relief valves (PSRVs)[40] "are capable of closing following discharge of subcooled water." Westinghouse noted that the PSRVs might have been designed or "qualified" to relieve subcooled water. Westinghouse also noted that "licensees may have qualified these valves in compliance to NUREG-0737, Item II.D.1." If the PSRVs were not designed or qualified for subcooled water discharge, Westinghouse recommended that licensees reevaluate the IOECCS event with three possible options of (1) reducing emergency core cooling system (ECCS) flow used in the safety analysis, (2) using a less restrictive operator response time, or (3) crediting the use of one or more PORVs to help mitigate the accident.

~~In~~ Later, in Supplement 1 to NSAL-93-013, Westinghouse alerted licensees to potential reduced time for operator action if a positive displacement pump (a typical ~~part~~ component of the CVCS) were in service, and to the need to qualify the PSRVs and the piping downstream of the PSRVs and PORVs if water discharge from the pressurizer is predicted.

Some licensees submitted license amendments that involved improvements to the PORVs and their circuitry to avoid water discharge through the PSVs (e.g., Salem[41], Millstone[42], Callaway[43], and Diablo Canyon[44]). The NRC staff review and approval of those proposed improvements relied on engineering judgment relative to the various test information and PORV circuitry upgrades described by individual licensees. The licensee for Byron and Braidwood submitted an LAR for similar PORV improvements,[45] but that request was later withdrawn.[46]

[39] NRC 1990a

[40] Westinghouse used the term PSRVs. The specific valves for Byron and Braidwood should be designated as "safety valves" or "pressurizer safety valves" as they are by the manufacturer, in the ASME BPV Code, and by the licensee. This difference in terminology is not significant to any of the findings or conclusions in this report.

[41] NRC 1997

[42] NRC 1998

[43] NRC 2000

[44] NRC 2004a

[45] ComEd 1998

[46] ComEd 1999

As indicated below, the Panel's sampling review found two plants, in addition to Byron and Braidwood, that chose to address this issue by crediting the capability of PSVs to relieve water, based on the EPRI testing performed in response to TMI Action Plan Item II.D.1.

3.6 Commission Paper on Passive Plant Designs (1994)

In 1994, in preparation for the design certification reviews of passive reactor designs (e.g., the Westinghouse Advanced Passive 1000 (AP1000) and the General Electric Economic Simplified Boiling-Water Reactor (ESBWR)), the NRC staff presented nine issues to the Commission for policy decisions.[47] Although PSV categorization and performance requirements were not explicitly addressed, the paper does include an issue on "Definition of Passive Failure" and an extensive discussion on whether check valves are passive or active components and how they should be addressed in current plants and future passive designs.

SECY-94-084 recognizesd the GDCs and SECY-77-439 as establishing long-standing requirements and guidance in this area. The paper acknowledgesd that the industry (including EPRI documents and ANSI/ANS 58.9[48]) have been inconsistent with respect to check valve failures, sometimes considering them as "active failures" and sometimes as "passive failures." In SECY-77-439, however, the NRC staff stated that the failure of a simple check valve to move to its correct position when required was a "passive failure." In addition, SECY-94-084 states that "[i]n licensing reviews, however, only on a long-term basis [e.g., long-term recirculation cooling following a loss of coolant accident (LOCA)] does the NRC staff consider passive failures in fluid systems as potential accident initiators in addition to initiating events." The paper also states that "[f]or current plants, the NRC staff normally treats check valves, except for those in containment isolation systems, as passive devices during transients or design-basis accidents."

Furthermore, SECY-94-084 states that "[r]edefining check valves as active components, subject to consideration for single active failures would cause these valves to be evaluated in a more stringent manner than that used in previous licensing reviews" (emphasis added). The NRC staff then recommended (and the Commission agreed[49]) that the NRC staff should "maintain the current licensing practice for passive component failures on the passive [advanced light water reactor] ALWR designs, and to redefine check valves, except for those whose proper function can be demonstrated and documented, in the passive safety systems as active components subject to single failure consideration." Therefore, the NRC's position on check valves was changed only for passive ALWR designs going forward.

The Panel considersed the opening function of check valves and PSVs to be similar in that they both open through the motion of the valve disk under differential pressure with no external signal or motive power. The Panel also recognizesed that the ambiguity with respect to "passive" versus "active" component definitions and nomenclature exists for safety valves. In addition, the passive or active classification of check valves or safety valves may differ based on design considerations, inservice testing, or accident analyses. For example, the PSVs and PORVs, as well as numerous check valves, are classified as active components in the Byron and Braidwood inservice testing programs. However, for purposes of applying the single failure criterion in the GDC context, the Panel concludesed that it is appropriate to consider the potential failure of a PSV following water discharge as a passive failure, consistent with the treatment of check valve failures for the operating fleet.

[47] NRC 1994a
[48] ANS 1981
[49] NRC 1994b

3.7 Draft Standard Review Plan Revision (1996)

The 1996 draft revision to SRP Sections 15.5.1 – 15.5.2 on IOECCS and CVCS malfunctions includes extensive updates to the 1981 revision, but neither version includes any discussion, criteria, or guidance on applying ASME Code requirements to PSVs or on applying the single failure criterion or any other failure assumption to PSVs.[50]

3.8 Power Uprate Reviews and License Amendments (2001-2006)

As part of the 2001 power uprate review for Byron and Braidwood, the NRC staff approved the analysis of an IOECCS (UFSAR Section 15.5.1) that included pressurizer filling, PSV water discharge, ECCS termination, and PSV closure. In the Backfit SE, the NRC staff indicate~~s~~d that the 2001 license amendment was predicated on the NRC's mistaken (unsubstantiated) belief that the valves were ASME-qualified (certified). However, ~~a~~ the Panel's review of the SE and associated RAIs show~~s~~ed that, in 2001, the NRC staff was well aware of the nature of the EPRI testing that the licensee relied on. The Panel did not find any evidence that the licensee claimed or the NRC staff believed that the valves were "qualified" in an ASME certification sense; rather, the record shows that the NRC staff thoroughly consider~~ation~~ed ~~of~~ the testing conducted on valves of the type installed at the plants and ~~a~~ applied well-informed and reasoned technical judgment in reaching its conclusion that ~~this~~ the EPRI testing provided appropriate qualification.

The Panel's understanding and conclusions about the staff review were ~~was~~ confirmed via discussions with the individual who was the responsible Section Chief in the Reactor Systems Branch at the time. He informed the Panel that the 2001 license amendment was based on the exercise of staff engineering judgment and that there was no discussion of ASME certification or qualification of valves. In addition, the Panel found that the NRC approved power uprates for other nuclear power plants that included comparable staff evaluations of water discharge through PORVs or PSVs based on test information provided by individual licensees. For example, in 2001, the NRC granted a power uprate for Shearon Harris that included the operability of PORVs and PSVs during the discharge of subcooled water, referencing TMI Action Plan Item II.D.1.[51] As noted above, in 2006, the NRC also granted a power uprate for Beaver Valley. The SE for this Beaver Valley amendment referred to RIS 2005-29 and ~~found~~ indicated that there was reasonable assurance that the PSVs would adequately discharge water and reseat following a spurious safety injection actuation, based on the EPRI test data from 1981 and an evaluation of the temperature of the liquid being discharged.

During the NRC evaluations of license amendments since the TMI-2 accident, the NRC staff has specified in some SEs that a PORV or PSV would be assumed to stick open if it was not qualified for liquid service. To address this concern, the NRC staff reviewed and accepted a variety of test information (including EPRI, Wyle, and vendor testing) submitted by individual licensees to demonstrate the capability of PORVs or PSVs to reseat following water discharge. In the sample of SEs it reviewed, the Panel did not find a specific requirement for the PORVs or PSVs to be certified under the ASME BPV Code as capable of passing water and reclosing.

In 2004, the NRC issued license amendments for Byron and Braidwood granting an adjustment to the PSV setpoints. In an RAI, the NRC staff requested that the licensee perform a quantitative analysis regarding the number of opening cycles during which the PSV would be expected to pass water and the temperature of the water being discharged. In the Setpoint SE,

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[50] NRC 1996
[51] NRC 2001d

the NRC staff concluded that the analysis was acceptable for assuring that the PSVs would remain operable following a spurious safety injection event.

3.9 RIS 2005-29 and Proposed Draft Revision 1 to RIS 2005-29 (2005)

In 2005, the NRC staff issued RIS 2005-29 "to notify licensees of a concern identified during recent reviews of power uprate [LARs]." The RIS addressed the manner in which some licensees acted in response to NSAL-93-013. The RIS was issued at the division level in NRR and does not include a record of office-level concurrence. The RIS was not reviewed by CRGR. Although requested by the Panel, the CRGR staff could find no documentation ~~was readily available~~ regarding the CRGR's decision not to review the proposed RIS before it was issued.~~, it~~ It appearsed to the Panel that the ~~lack of a~~ CRGR may not have reviewed ~~stemmed from~~ the RIS because of assertions in the RIS such as these:

- "This RIS requires no action or written response and, therefore, is not a backfit under 10 CFR 50.109. Consequently, the NRC staff did not perform a backfit analysis."
- "This RIS is informational and pertains to a NRC staff position that does not depart from current regulatory requirements and practice."

A key statement in RIS 2005-29 is the following (with emphasis added):

> The NRC staff's position is noted in the power uprate review standard, as follows: "For the [IOECCS] and [CVCS] malfunctions that increase reactor coolant inventory events: (a) non-safety-grade pressure-operated relief valves should not be credited for event mitigation and (b) pressurizer level should not be allowed to reach a pressurizer water-solid condition.".

However, the ~~cited~~ NRC staff review standard cited in the RIS (RS-001)~~, which~~ is explicitly limited to EPU reviews, ~~states~~ stating that "[t]he staff does not intend to impose the criteria and/or guidance in this review standard on plants whose design bases do not include these criteria and/or guidance. No backfitting is intended or approved in connection with the issuance of this review standard."[52]

This intent of RS-001 to define and clarify the scope of EPU reviews, but not impose new requirements or new interpretations of requirements, was confirmed by the Panel in discussions with the manager responsible for developing and issuing RS-001. Therefore, contrary to the RIS statement, neither RS-001 nor RIS 2005-29 documented "known and established standards of the Commission" applicable to Byron and Braidwood.

The Panel also notes that neither RIS 2005-29 nor its draft Revision 1,[53] which is currently under development, discuss water discharge certification requirements in accordance with the ASME BPV Code. In fact, as stated above, the NRC issued a 2006 power uprate amendment for Beaver Valley in which the SE cited RIS 2005-29 and yet relied on the EPRI testing data to address the concern.

3.10

[52] NRC 2003
[53] NRC 2015a

3.10 SECY-05-0138 (2005)

SECY-05-0138 presents a comprehensive history of the application of the single failure criterion, including extensive discussion of the treatment of passive components in fluid systems.[54] The paper enclosed a July 2005 draft of a staff technical report on the single failure criterion. Section 4.2.2 of this report acknowledges that "[o]ne particular issue identified in this project is the continued existence of the footnote to the definition of single failure in 10 CFR [Part] 50 Appendix A stating that the regulatory position on considering passive failures in fluid systems is under development." In Section 2.5.3, the draft report quotes from SECY-77-439 (discussed above) and recognizes that in current practice, as in 1977, "[p]assive failures in fluid systems are generally excluded from single-failure assessments."

SECY-05-0138 and the accompanying draft report present three alternatives for using a risk-informed and performance-based approach to address the single failure issue. The draft report clarifies that all of the alternatives "could include developing a position on single passive failures in fluid systems to replace the footnote now in 10 CFR Part 50 Appendix A definitions."

These documents make it clear that, with few exceptions, neither the NRC staff nor the Commission has established specific requirements relating to the treatment of passive component failures in fluid systems. The Panel believes the existence of this Commission paper, contemporaneous with discussions on potential PSV failures (e.g., RIS 2005-29), makes it clear that no specific "known and established standards" on PSV failures had been developed between 1977 and the time of the Byron and Braidwood license amendments in 2001 and 2004.

3.11 Standard Review Plan Revision (2007)

Revision 2 to SRP Section 15.5.1 states that "[t]he pressurizer safety valves, too, may be assumed to reseat properly after having relieved water; but only if such valves have been qualified for water relief." However, this section does not reference ASME BPV Code requirements for safety valve certification.

3.12 Backfit Letter and Subsequent Backfit Appeals (2015-2016)

The Backfit SE is predicated on the following positions:

- "water relief through a valve that is not qualified for water relief will cause that valve to stick in its fully open position" (emphasis added)
- "the licensee … has not applied the single-failure assumption" (emphasis added)
- "nor ~~have they~~ [has the licensee] provided ASME water qualification documentation for the PSVs … the ASME … original Overpressure Protection Report … inservice test history … including both water and steam tests" (emphasis added)

The Backfit SE ~~argues~~ contends that an IOECCS would escalate to a more severe event. Such an escalation would be contrary to the Byron and Braidwood licensing basis (i.e., contrary to the ANS non-escalation position) and could be in non-compliance with the GDCs (as included in the Byron and Braidwood licensing basis) since an IOECCS with a stuck-open valve had not been analyzed and shown to meet the appropriate criteria for an AOO.

[54] NRC 2005a

Based on its review of all the relevant documents and discussions with the individuals (staff and managers) involved in the original review and the backfit, the Panel has developed an understanding of the regulatory requirements and practices, the potential safety issues, and backfit rule obligations. The Panel has determined that the numerous, complex, and detailed regulatory and technical issues all depend on the answers to two critical questions on valve performance:

- Must the PSVs in question be assumed to fail given liquid water discharge because of the lack of ASME certification for water discharge?
- Must the PSVs be assumed to fail in accordance with the GDC "single failure" requirements?

In the Backfit SE, the NRC staff indicate~~s~~d that "[o]ne assumption that is particularly important to the non-escalation criteria is that water relief through a valve that is not qualified for water relief will cause that valve to stick in its fully open position" (emphasis added). The Panel conclude~~s~~d that this issue—the treatment of potential valve failure—is not only "particularly important," it is the critical issue upon which the compliance backfit hinges.

Based on the historical evidence, the Panel conclude~~s~~d that there is not now, nor has there been, a known and established Commission standard (1) that PSVs must be assumed to fail following water discharge in the absence of ASME certification for water discharge, or (2) that PSVs must be assumed to fail as part of single failure criterion analysis. The NRC staff's determination that ASME certification is necessary first appears in the Backfit SE. The determination that application of the single failure criterion is necessary first appears in the draft Revision 1 to RIS 2005-29. The Panel has not identified these positions being stated in any final NRC requirement or guidance document.

The Panel also conclude~~s~~d that in 2001 and 2004 and at present, the known and established standard of the Commission is 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. In preparing the Uprate SE and the Setpoint SE, the NRC staff exercised reasonable and well-informed engineering judgment when the NRC staff concluded that the PSVs were unlikely to stick open. ~~The~~ On the bases of it document reviews and interviews, the Panel concluded that the NRC staff reviewers involved in the 2001 power uprate review were among the most experienced and senior reviewers in their areas of expertise. The NRC staff valve expert involved in the review was the agency's most knowledgeable individual on PSVs and the relevant ASME Code requirements, and was a nationally recognized expert. The Panel did not find any evidence that the NRC staff's issuance of the 2001 or 2004 license amendments was based on an omission or mistake of fact. Rather, the Panel concluded that the current NRC staff positions on valve qualification in the Backfit SE are new or modified interpretations of compliance.

In interactions with the Panel, NRR staff emphasized several issues raised in the Backfit Letter. The Panel summarizes its consideration of those issues in the following subsections.

3.12.1 Non-Escalation Position and Valve Failure

In the Backfit SE, the NRC staff discussed the definition of event conditions in ANS-51.1/N18.2-1973 and the provision in this standard that events of one condition do not propagate to cause a more serious fault. ~~(~~This position is commonly known as the non-escalation position~~)~~. In

~~interactions~~ a meeting with the Panel, NRR staff provided several clarifications on this topic, summarized by the Panel as follows:

- ANS-51.1/N18.2-1973 defines the categories of design basis transients and accidents based on an anticipated frequency of occurrence (annually for ANS Condition II events).
- It is a long-standing NRC position that escalation from one condition to another is not acceptable.
- ANS-51.1/N18.2-1973 constitutes a known and established standard that has been reflected in NRC guidance documents and in the licensing basis of each U.S. nuclear power plant.

The Panel confirmed that this ANS standard is referenced in several places in Chapter 15 of the Byron and Braidwood UFSAR. The Panel agrees that the non-escalation position is an established standard applicable to Byron and Braidwood, but did not identify historical evidence that implementation of this standard requires Exelon to assume that its pressurizer valves will fail open under water discharge conditions, to apply the single failure criterion to PSV failure in these circumstances, or to impose ASME Code requirements for certification, qualification, or testing of PSVs for water discharge.

3.12.2 Non-Escalation Position and Return to Service

In the Backfit SE, the NRC staff makes reference to the time it would take to clean up a contaminated containment following a stuck-open pressurizer valve. In ~~interactions~~ its discussions with the Panel, NRR staff re-emphasized concerns that extended steam and water discharge through the pressurizer valves would result in the failure of the pressurizer relief tank rupture disk, would require repair of the damaged PSVs, and might cause an extended time period for the return to service of the nuclear power plant.

The Panel does not consider the time period necessary for the licensee to perform radioactive clean-up activities in the containment building, to inspect and conduct any necessary repairs to the PSVs, or to prepare for plant startup, to constitute issues that support a compliance backfit imposed by the NRC. The NRC staff and inspectors would verify that the licensee would conduct these activities ~~are conducted~~ as appropriately to protect the public health and safety prior to plant restart. The Backfit SE states that UFSAR Section 15.5.1.3 "implie[s]" that the plant will return to operation in a "short period," but the Panel found no bases ~~sees no support~~ for a timing requirement in UFSAR Section 15.5.1.3. Also, the Panel ~~has~~ did not ~~identified~~ find a regulatory ~~interest~~ requirement or basis for ~~in~~ defining or limiting the time ~~needed~~ available for the plant to return to operation.

3.12.3 TMI Action Plan Item II.D.1 and EPRI Testing

Although the Backfit Letter and NRR Appeal Decision do not speak explicitly to TMI Action Plan Item II.D.I, in interactions with the Panel, NRR staff stated that the known and established standard in question is the TMI Action Plan Item II.D.I standard for licensees and applicants to conduct testing to qualify the RCS relief and safety valves under expected operating conditions for design-basis transients and accidents. As discussed above and in Appendix B to this report, the NRC accepted the EPRI testing to satisfy TMI Action Plan Item II.D.1 for Byron and Braidwood in SEs forwarded by letters in 1988 and 1990. Therefore, the Panel ~~considers~~ conclude that this known and established standard referenced by the NRC staff ~~to have~~ had been met for Byron and Braidwood.

In interactions with the Panel, the NRR staff further stated that an omission or mistake of fact occurred when the licensee failed to acknowledge that the EPRI testing program did not evaluate water discharge from the pressurizer valves during extended high pressure safety injection for Byron and Braidwood. As discussed in Appendix B to this report, in the 1988 and 1990 SEs ~~on~~ for the Byron and Braidwood responses to TMI Action Plan Item II.D.1, the NRC staff evaluated the capability of the PSVs and PORVs during feedwater line break accidents, including water discharge. In these SEs, the NRC staff found that the performance of the PSVs and PORVs with water discharge was acceptable based on the EPRI tests. Therefore, the Panel ~~does not agree~~ also concluded that the licensee's reference to the EPRI testing program was not an omission or a mistake of fact.

3.12.4 ASME Code Certification

In the Backfit SE, the NRC staff stated that certain ASME Code information would be necessary to support water qualification of the PSVs. In ~~interactions~~ its meeting with the Panel, NRR staff stated that, to satisfy the standard for water discharge capability of pressurizer valves, it would be necessary to conduct flow capacity certification in accordance with the ASME BPV Code and inservice testing throughout the service life in accordance with the ASME OM Code. The NRR staff referenced certain licensing actions in which water discharge was not considered acceptable, or different actions were required.[55]

As discussed in Appendix C to this report, the NRC staff required additional actions for some licensees to support reliance on the PORVs for water discharge and to avoid water discharge through the PSVs. The Panel found, however, that the NRC staff also allowed some licensees to rely only on EPRI testing without significant additional activities. The Panel did not identify instances where the NRC staff imposed certification by the ASME BPV Code and testing in accordance with the OM Code, or required alternatives to the ASME BPV or OM Codes, in the examples of NRC staff review of water discharge capability for pressurizer valves.

~~In interactions with the Panel, the~~ The NRR staff also identified for the Panel specific ASME Code provisions that it viewed as supporting ~~the~~ its position that ASME Code requirements apply to qualification of pressurizer valves for water discharge. The NRR staff, however, did not provide evidence that ~~these provisions have~~ the staff has consistently ~~been~~ interpreted these positions as the NRC staff is now interpreting them. Given the inconsistencies in the NRC staff's ~~treatment~~ handling of TMI Action Plan Item II.D.1 and ~~the NRC staff's historical~~ variations in its licensing practices, the Panel conclude~~s~~d that the NRR staff's current application of the ASME Code is not supported by the historical record.

3.12.5 Conduct of 2001 and 2004 License Amendment Reviews

In light of the wide range of positions taken by the NRC staff ~~positions~~ during ~~the~~ its reviews of pressurizer valve capability since the TMI-2 accident, the Panel agrees that, in the course of preparing the 2001 Uprate SE or Setpoint SE, the NRC staff could have considered the need for the licensee for Byron and Braidwood to improve the reliability of the PSVs or PORVs for water discharge or to avoid water discharge through the PSVs by PORV improvements. The NRC staff may have been able to justify additional actions, but they determined that it was not necessary. Instead, the NRC staff reviewers in 2001 used their expert engineering judgement to determine that it was not necessary to assume that the PSVs or PORVs would stick open with

[55] Salem (NRC 1997), Millstone (NRC 1998), and Callaway (NRC 2000)

water discharge, based on EPRI test information, licensee supplemental information, and their own technical experience.

In discussions with the Panel, NRR staff raised a concern that the Setpoint SE does not document a re-review of the qualification of the PSVs and noted that if the Uprate SE had not found water discharge through the PSVs to be acceptable, it is unlikely that the NRC staff would have approved this 2004 amendment. In Appendix C to this report, the Panel summarizes the discussion in the Setpoint SE of the PSV water discharge capability. The Panel recognizes that a staff review may rely on a previous more extensive review to determine the acceptability of a similar request. The Panel does not consider the review approach used in 2004 to challenge the ~~adequacy~~ acceptability of the 2001 review.

4 RESPONSE TO THE EDO QUESTIONS

In establishing the Panel, the EDO asked the Panel to answer five specific questions, as well as evaluating the overall appropriateness of the backfit. The Panel's answers to these questions are provided below.

4.1 Were the approvals based on a mistake? If so, what was the mistake and what are the implications for Byron and Braidwood?

In responding the question, the Panel has considered the differing views of the NRR staff and the licensee on this issue. Those positions are summarized below:

- In the NRR Appeal Decision, the NRC staff claim~~s~~ed that "[t]he NRC erred in approving a sequence of events that allowed the [IOECCS], [CVCS] malfunction, and inadvertent opening of a pressurizer safety or relief valve analyses in the 2001 and 2004 [SEs]" and "the NRC staff understood the PSVs to be qualified for water relief when, in fact, they were not."
- Exelon claims in the NRR Backfit Appeal that "the compliance exception requires more than simply asserting that the prior staff approvals were wrong—the NRC must demonstrate that the prior approvals were erroneous because of an omission or mistake of fact at the time of the approval. The NRC has not made that case here."

~~The~~ On the basis of its independent review, the Panel conclude~~s~~d that, in 2001 and 2004, the NRC staff did not misunderstand the qualification status of the PSVs and that it was not a mistake to undertake a review of or make a technically based safety finding on the likely successful performance of the valves. In the Panel's opinion, the actions of the Reactor Systems Branch in 2001 to reach out to the Division of Engineering's Mechanical Engineering Branch for expert technical review assistance was both appropriate and commendable. The NRC staff reviewers involved in the 2001 power uprate review were among the most experienced and senior reviewers in their areas of expertise. The valve expert involved in the review was the NRC's most knowledgeable individual on PSVs and the relevant ASME Code requirements, and was a nationally recognized expert. ~~The Panel cannot agree that the NRC staff was misinformed, ill-informed, or in error, or that it made incorrect or inappropriate decisions~~ For these reasons, the Panel concluded that the NRC staff reviews and approvals of the 2001 and 2004 license amendments were not based on omissions or mistakes of fact.

Comment [SW]: See my comment, below.

Comment [SW]: Our answer is based largely on the reputation of the past reviewers. While I agree that this is important and should be included in our answer, it does not address the highlighted statement, above. I think we need to be a bit more precise in the first part of the answer and say that the valves were not required to be qualified for water relief and that the 2001 staff knew that and knew what they were approving. We can use the words we already have in our report to do this. I also think the tone of our answer may be too harsh and could come across as overly critical of today's staff (in a report this likely to be made publicly available). (Not to mention that its directed to the EDO.) As we learned through our total emersion, this is a complex issue that has been handled in a variety of ways over time. Our guys just happened to pick the wrong way this time (according to us). As a minimum I suggest we tone down our response by deleting the underlining and revising the last sentence as shown. We may also want to convince ourselves that we are not inadvertently (or unfairly) comparing the expertise and capabilities of the past staff with that of the current staff.

4.2 What is the known and established standard for water qualification of PSVs?

The Panel conclude~~s~~d 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. ~~No~~ The Commission has not established a more detailed or prescriptive standard ~~has been promulgated by the Commission~~.

4.3 What is the known and established standard for progression of postulated events between categories of severity?

For Byron and Braidwood, the NRC staff and the Panel agreed that the known and established standard for progression of postulated events between categories of severity is the so called non-escalation position specified in ANS-51.1/N18.2-1973. This position, which is ~~set forth~~ included in the Byron and Braidwood UFSARs, requires that events of one condition do not propagate to cause a more serious condition ~~as described above. The Panel supports the NRC staff's view that non-escalation~~ (i.e., from ANS Condition II to ANS Condition III or IV) ~~is a known and established standard applicable to Byron and Braidwood. This issue of event escalation is also a focus of RIS 2005-29 and the draft Revision 1 to RIS 2005-29 that was issued for public comment in 2015.~~ The Panel conclude~~s~~d that the IOECCS (an AOO per the GDC definition and an ANS Condition II event) would escalate to a more severe event if a PSV were to stick open, or if both a PORV stuck open and its block valve failed to close. Such an escalation would be contrary to the Byron and Braidwood licensing basis (i.e., contrary to the ANS non-escalation position) and could be in non-compliance with the GDC (as included in the Byron and Braidwood licensing basis), since an IOECCS with a stuck-open valve had not been analyzed and shown to meet to appropriate criteria for an AOO. However, this event progression standard does not establish specific standards for valve qualification to determine whether a valve would stick open and cause this escalation. Therefore, the Panel concluded that it is not the basis for a compliance backfit given the current set of facts. (Additional information about ANS-51.1/N18.2-1973, is included in Section 13.12.1 of this report.)

4.4 Does the current licensing basis for Byron and Braidwood comply with the applicable regulations? Is it adequate to provide protection to public health and safety?

The Panel conclude~~s~~d that the current licensing basis for Byron and Braidwood ~~do~~ comply with the applicable regulations based on the UFSAR analyses, which the NRC staff found acceptable through a reasonable and technically sound evaluation using appropriate Commission safety standards.

4.5 Given that Exelon suggests that the NRC pursue a cost-justified substantial safety enhancement backfit, what is the contribution to overall plant risk of the current configuration at Byron and Braidwood?

The Panel requested RES to provide information and insights on the risk significance of the sequence at issue, to assure that the Panel's judgments were being made with a full understanding of their significance, and to assist in responding to the EDO question.

The RES study suggests that the most significant IOECCS sequence, assuming that all pressurizer overfill events lead to a small LOCA, contributes approximately 1 percent of the total internal event core damage frequency (CDF). In its report, RES estimated a that the maximum

Comment [SW]: Not sure this adds anything to the answer.

Comment [SW]: We didn't answer this question. We, of course, have no idea if Byron and Braidwood are meeting their licensing basis, but the licensing basis itself, based on the staff's SER, should, I guess, be adequate to provide protection of public health and safety. The more important point is that compliance with the backfit is not needed to provide adequate protection.

Comment [SW]: Is this report included as a reference? Can it be made public?

benefit (CDF reduction) ~~from a "perfect backfit" (i.e., always preventing pressurizer overfill)~~ of 1.5E-07 per year would be achieved if the plants were modified (backfit) such that pressurizer overfilling was always prevented. If the PSVs are not assumed to always fail following water discharge (consistent with the NRC staff expert judgment in 2001~~) or a smaller improvement than a "perfect backfit" were considered~~, the risk-reduction benefit of implementing the backfit would be even smaller.

Comment [SW]: I'm not sure I did it, but I suggest we find a way to make the point with using an undefined term.

The Panel is aware of and sensitive to two important issues related to this question. First, NRR, not the appeal Panel, is responsible for any decisions on alternative application of the backfit rule to this issue (through the other categories of adequate protection or cost-justified substantial safety enhancement). Second, the Panel does not wish to imply that "the contribution to plant risk" should be seen as the only measure of enhanced safety. The issues of event classification and the non-escalation of events are essentially defense-in-depth concepts. Defense in depth has a recognized role and value in the regulatory process. The Panel is also aware that not every defense-in-depth feature has the same safety significance, and that the estimated risk significance (measured in core damage frequency) is very relevant.

Within the context described above, the Panel concluded~~s~~ that the contribution to overall plant risk is very small.

5 SUMMARY AND CONCLUSIONS

The compliance exception to the Backfit Rule is intended to address failures to meet known and established Commission standards because of omission or mistake of fact. New or modified interpretations of what constitutes compliance do not fall within the exception. Therefore, to address the appeal of the proposed compliance backfit, the Panel focused on determining if this case is most appropriately characterized as one in which the licensee "failed to meet known and established standards of the Commission because of omission or mistake of fact," or rather as a case of a "new or modified interpretations of what constitutes compliance."

The NRC staff's compliance backfit argument depends on two separate determinations:

1. the assumed failure of PSVs to reclose after passing water, and

2. the necessity of preventing "event escalation" (i.e., the position that "an incident of moderate frequency should not generate a more serious plant condition without other faults occurring independently").

For the NRC staff's compliance backfit conclusion to be valid, both of these determinations must meet the above compliance backfit standard by involving failure to meet known and established standards of the Commission.

In the first of these determinations, the NRC staff's compliance backfit is based on the assumption in the Backfit SE that the PSV fails to reclose given the absence of "ASME water qualification documentation." As indicated in the Backfit SE, the Uprate SE involved a technical evaluation of safety valve capability and likely performance under water-discharge conditions rather than a simple assumption of a failure. The NRR Appeal Decision indicates that "the 2001 and 2004 [license amendment] approvals occurred because the NRC staff understood the PSVs to be qualified for water relief when, in fact, they were not."

The Panel carefully considered these views and has reviewed the relevant documents including the licensee's responses to the NRC staff's RAIs,[56] the NRR technical branch's SE input,[57] and the Uprate SE. 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 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.

On the basis of its review, the Panel concluded that the NRC staff who prepared the Uprate SE did not misunderstand the qualification status of the PSVs and that it was not a mistake to undertake a review of or make a technically based safety finding on the likely successful performance of the valves. In the Panel's opinion, the actions of the Reactor Systems Branch in 2001 to reach out to the Division of Engineering's Mechanical Engineering Branch for expert technical review assistance was both appropriate and commendable. The NRC staff reviewers involved in the 2001 power uprate review were among the most experienced and senior reviewers in their areas of expertise. The NRC staff valve expert involved in the review was the agency's most knowledgeable individual on PSVs and the relevant ASME Code requirements, and was a nationally recognized expert. The Panel disagreed that the NRC staff was misinformed, ill-informed, or in error, or that it made incorrect or inappropriate decisions.

Comment [SW]: Same comments as above for our response to Question 1.

The Panel concluded that three related technical and regulatory positions related to the PSVs (separate from the issue of the non-escalation position) underpin the backfit:

1. ASME water qualification (certification) documentation is required if a valve is to be assumed to reclose after passing water.

2. Water discharge through a steam-qualified valve will cause that valve to stick in its fully open position.

3. PSVs are subject to a single-failure assumption.

~~None~~ In the Panel's view, none of these three positions were "known and established standards of the Commission" in 2001 or 2004 for determining when it was appropriate to assume a failure of PSVs to reseat. In fact, they were not "known and established standards of the Commission" in 2005 (when RIS 2005-29 was issued) or 2006 (when the Beaver Valley EPU was approved) or 2007 (when Revision 2 to SRP Section 15.1.1 – 15.1.2 was issued).

Moreover, these positions do not appear to be "established standards of the Commission" at present. The 2007 version of SRP Section 15.1.1 – 15.1.2 allows credit for PORVs and PSVs if they have been "qualified for water relief." The NRC staff's determination that ASME certification is necessary first appears in the Backfit SE and is not addressed in any of the NRC requirements or guidance documents reviewed by the Panel. The determination that application of the single failure criterion is necessary first appears in the draft Revision 1 to RIS 2005-29, which is still in process, and is not included in any NRC requirement or guidance document reviewed by the Panel.

The Panel 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. In earlier documents addressing this topic, beginning with NUREG-0737, it is the Panel's view that the use of the

[56] ComEd 2000b, Exelon 2001
[57] NRC 2001a

word "qualified" or "qualification" implies a general demonstration of capability, such as in the EPRI testing done in response to TMI Action Plan Item II.D.1. In light of this standard, the Panel concluded that, when preparing the Uprate SE and the Setpoint SE, the NRC staff exercised reasonable and well-informed engineering judgment to conclude that the PSVs were unlikely to stick open.

Overall, the Panel concluded that the NRC staff's position on valve qualification in the Backfit SE is a new or modified interpretation of what constitutes compliance in addressing potential PSV failures following water discharge. Although this new staff position represents a well-intentioned and conservative approach that could provide additional safety margin, the Panel concluded that it does not provide a basis for a compliance backfit.

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.

The Panel's findings, therefore, support the Exelon backfit appeal.

6 ADDITIONAL PANEL THOUGHTS

In addition to the specific finding relating to the backfit appeal, the Panel believes it is important to acknowledge, and for the NRC staff and licensees to appreciate, that water discharge through a PSV not specifically designed for such service is undesirable and should be minimized or avoided as a matter of conservative engineering and prudent operations. This is reinforced by the information provided in NSAL-93-013 and its Supplement 1, and the actions by various licensees in response to these documents, as well as the limited scope of the EPRI testing conducted over 30 years ago.

Operator training, control room procedures to terminate the event before pressurizer filling, and use of PORVs rather than reliance on PSVs, are clearly preferred and prudent measures, whether they form the facilities' UFSAR licensing basis and are assumed in the accident analyses or not.

The PSVs in question were designed for steam service. Steam relief is their normal service condition and applies to their ASME BPV Code certification. The Panel supports the previous NRC staff determinations for Byron and Braidwood and certain other plants that PSVs experiencing water discharge during an abnormal or accident condition need not be assumed to fail since there was a reasonable and technically well-informed engineering judgement to the contrary. However, the Panel also considers the actions by various licensees to improve the reliability and performance of the PORVs to avoid water discharge through the PSVs to be prudent in light of the design specifications of the PSVs.

The Panel considered but could not determine the extent to which the licensee for Byron and Braidwood addressed crediting water discharge through the PSVs, PORVs, or PORV block valves in the Byron and Braidwood inservice testing programs. The Panel recognizes that the difference between the intended use of these valves for overpressure protection and their infrequent use in response to certain plant events might be considered in implementing appropriate inservice testing activities.

The Panel notes that water discharge through various pressurizer valves is not a new issue because water discharge has always been credited (by the licensee for Byron and Braidwood and other licensees) for the feedwater line break analysis in UFSAR Section 15.2.8.

On the basis of its review, the Panel also noted, as did a member of the earlier NRR backfit appeal panel (ADAMS Accession No. ML16081A405), that the issue of pressurizer valve performance following water discharge appears to have generic applicability, and is not specific to only Byron and Braidwood. The Panel believes that resolution of this issue would have benefited from consideration of the generic nature of the issue through the appropriate NRC processes. The Panel included the information it gathered and assessed to reach its conclusion regarding the generic nature of the issue in Appendices B and C of this report. Should the NRC staff undertake a generic look of the issues, it should, among other things, consider the information presented and questions raised in those appendices. The review should also include a reassessment of the information and staff positions communicated in RIS 2005-29, as well as those included in its proposed Revision 1, which is currently in process, to determine whether or not the RISs include new staff positions with the potential for inappropriate or unintended backfitting. As part of any generic assessment, the Panel also recommends that staff determine if the information in RIS 2005-29 and its proposed Revision 1 should be incorporated into a regulatory guide or another guidance document.

Comment [SW]: I suggest we add something like this to the body of the report. (I lifted the beginning of it from the cover memo.)

Comment [SW]: If Tony's memo is not publicly available, we can probably strike this from the report and just go with our own thoughts on this matter.

APPENDIX A: HISTORY OF THE BACKFIT RULE AND THE COMPLIANCE EXCEPTION

The Backfit Rule

Title 10 of the *Code of Federal Regulations* (10 CFR), Section 50.109, "Backfitting," was originally promulgated in 1970.[58] Because of perceived deficiencies in the rule, the U.S. Nuclear Regulatory Commission (NRC) substantially revised it in 1985.[59] The 1985 rule was challenged in court, and the U.S. Circuit Court for the District of Columbia (D.C. Circuit) vacated this rule in its entirety. The D.C. Circuit took this action because it concluded that the revised rule could be interpreted to allow the NRC to consider costs in defining or redefining what is required for adequate protection of the public health and safety.[60] In response, the NRC revised the Backfit Rule in 1988 to remove any implication that costs could be considered in defining or redefining adequate protection.[61] The 1988 revisions only differed from the 1985 rule to the extent necessary to address the court's concerns. The 1988 rule was also challenged in court, but this time the D.C. Circuit upheld the rule.[62]

In its current form, 10 CFR 50.109(a)(1) defines backfitting as

> … the modification of or addition to systems, structures, components, or design of a facility; or the design approval or manufacturing license for a facility; or the procedures or organization required to design, construct or operate a facility; any of which may result from a new or amended provision in the Commission's regulations or the imposition of a regulatory staff position interpreting the Commission's regulations that is either new or different from a previously applicable staff position … .

Unless one of three specified exceptions apply, the NRC may impose a backfit only if it performs a backfit analysis in accordance with 10 CFR 50.109(a)(2) and determines in accordance with 10 CFR 50.109(a)(3) "that there is a substantial increase in the overall protection of the public health and safety or the common defense and security to be derived from the backfit and that the direct and indirect costs of implementation for that facility are justified in view of this increased protection."

Section 50.109(a)(4) sets forth the three exceptions to the requirements of 10 CFR 50.109(a)(2) and (a)(3). The first exception, the compliance exception, applies if the "modification is necessary to bring a facility into compliance with a license or the rules or orders of the Commission, or into conformance with written commitments by the licensee." 10 CFR 50.109(a)(4)(i). The second and third exceptions relate to actions ensuring adequate protection or to actions that involve defining or redefining adequate protection. 10 CFR 50.109(a)(4)(ii)-(iii).

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[58] AEC 1970 (Author and year citations in footnotes refer to the designation of references in Appendix D to this report.)
[59] NRC 1985
[60] *Union of Concerned Scientists v. U.S. Nuclear Regulatory Com'n*, 824 F.2d 108, 119-20 (1987).
[61] NRC 1988b
[62] *Union of Concerned Scientists v. U.S. Nuclear Regulatory Com'n*, 880 F.2d 552 (1989).

Commission Policy

The Commission addressed its intended application of the compliance exception in the 1985 rulemaking:[63]

> The compliance exception is intended to address situations in which the licensee has failed to meet known and established standards of the Commission because of omission or mistake of fact. It should be noted that new or modified interpretations of what constitutes compliance would not fall within the exception and would require a backfit analysis and application of the standard.

In the 1985 rule, the Commission acknowledged that staff interpretations of regulations are not legally binding, but the Commission also stated that "staff interpretations of broadly stated rules are often necessary to give a rule effect and in some instances may be a causal factor in initiating a backfit."[64] The Commission also stated, "Many of the most important changes in plant design, construction, operation, organization, and training have been put in place at a level of detail that is expressed in staff guidance documents which interpret the intent of broad, generally worked [sic] regulations."[65]

Backfitting Guidance

Extensive information regarding the appropriate implementation of backfitting is provided in the NUREG-1409.[66] Relevant excerpts from this guidance are provided below.

Applicable Regulatory Staff Positions

According to NUREG-1409, to be a backfit, "a new or revised staff position or requirement must be involved, that is, there must be a change in content or applicability of the previously applicable regulatory staff position (in the direction of increased safety requirements) … ." An applicable regulatory staff position is a requirement or position already specifically imposed on or committed to by a licensee. Examples of applicable regulatory staff positions include:

- legal requirements, as in explicit regulations, orders, and plant licenses and in amendments, conditions, and technical specifications
- written licensee commitments such as those contained in the final safety analysis report, licensee event reports, and docketed correspondence, including responses to NRC bulletins, generic letters, inspection reports, or notices of violation and confirmatory action letters
- NRC staff positions that are documented explicit interpretations of more general regulations and are contained in documents such as the Standard Review Plan, branch technical positions, regulatory guides, generic letters, and bulletins

A similar list of examples is provided in Manual Chapter 0514[67], which is also included as Appendix D to NUREG-1409. Manual Chapter 0514 was referenced in the 1988 rulemaking,

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[63] NRC 1985, at 38103
[64] *Id.* at 38102
[65] *Id.* at 38103. The 1988 rulemaking neither revised the compliance exception as stated in the 1985 rule nor provided additional guidance on its interpretation.
[66] NRC 1990c

and a working draft was provided to the Commission for information in SECY-88-102.[68] Manual Chapter 0514 provides a definition of “applicable regulatory staff positions” that is slightly more detailed than the definition in NUREG-1409. This definition from Manual Chapter 0514 is quoted below, with additional detail beyond NUREG-1409 emphasized in underlined text.

> Applicable regulatory staff positions are those already specifically imposed upon or committed to by a licensee at the time of the identification of a plant-specific backfit, and are of several different types and sources:
>
> a. Legal requirements such as in explicit regulations, orders, plant licenses (amendments, conditions, technical specifications). Note that some regulations have update features built in, as for example, 10 CFR 50.55a, Codes and Standards. Such update requirements are applicable as described in the regulation.
>
> b. Written commitments such as contained in the [Final Safety Analysis Report], [Licensee Event Reports], and docketed correspondence, including responses to Bulletins, responses to Generic Letters, Confirmatory Action Letters, responses to Inspection Reports, or responses to Notices of Violation.
>
> c. NRC staff positions[69] that are documented, approved, explicit interpretations of the more general regulations, and are contained in documents such as the [Standard Review Plan], Branch Technical Positions, Regulatory Guides, Generic Letters, and Bulletins; and to which a licensee or an applicant has previously committed to or relied upon. Positions contained in these documents are not considered applicable staff positions to the extent that staff has, in a previous licensing or inspection action, tacitly or explicitly excepted the licensee from part or all of the position.[70]

How Regulatory Positions are Established

NUREG-1409 provides responses to a number of questions regarding backfitting. The following response was given to questions asking, “Is it appropriate for the NRC staff to rely on informal or formal communications to other licensees as official NRC positions? What about NRC tacit approval of documents?”

> Informal or formal communications to one licensee are not official positions to all licensees. Section 053 of Manual Chapter 0514 identifies what can be applied as official staff positions in a plant-specific context. They are legal requirements such as contained in explicit regulations, orders, and plant licenses; written commitments such as contained in final safety analysis reports, licenses event reports, and docketed correspondence; and documented, approved explicit interpretations such as contained in the [Standard Review Plan], branch technical positions, regulatory guides, generic letters, and bulletins. Orders, licenses, and written commitments are applicable only to a particular licensee.

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[67] NRC 1988c
[68] NRC 1988a
[69] Requirements may be imposed by rule or order. Staff interpretations such as examples of acceptable ways to meet requirements are not requirements in and of themselves.
[70] Imposition of a staff position from which a licensee has previously been excepted is a backfit.

> If the NRC staff previously exempted a licensee from a legal requirement or approved position, it is not applicable to that licensee for the purpose of backfit consideration. Explicit exemption would be done formally in writing. The Appendix to NRC Manual Chapter 0514 discusses tacit approval under reanalysis of issues. Two situations are covered. In the first case, staff review of a previously accepted licensee action or program may result in a requested change. This would be classified as a backfit because it represents a change in a previous staff position and would require a backfit analysis (or a documented evaluation if it meets one of the exceptions listed in the backfit rule). In the second case, a licensee submittal committing to a specific course of action that has not received timely NRC staff review is implemented by the licensee. In this case, it is considered that the NRC staff tacitly accepted the licensee's action since timely notice to the contrary was not given. If the NRC staff subsequently adopts a different position and requests a change in the licensee action, this change may be classified as a backfit and thus require a backfit analysis (or a documented evaluation if it meets one of the exceptions listed in the backfit rule).

NUREG-1409 also addresses a question regarding tacit approvals by an inspector: "If an inspector has previously accepted (i.e., provided tacit approval of) a licensee's method, does a specific request for change constitute a backfit and if so, is a backfit analysis required?" The response is:

> Cases where an inspector provides tacit approval are relatively rare. Simply not challenging a licensee's practice normally would not be considered tacit approval. The only example provided in Manual Chapter 0514 is a case where the NRC has indicated tacit approval by not acting in a reasonable time on a licensee submittal and the licensee has moved ahead to implement the proposal described in the submittal. For the purpose of this question, it would most likely arise in connection with review of a licensee response to an inspection report.
>
> Explicit approval could be provided in an inspection report that states that a particular approach is acceptable. However, conclusions of that nature are usually made in [safety evaluations] rather than inspection reports.

Compliance Backfit Guidance

NUREG-1409 gives the following response to the question, "[h]ow does the backfit rule apply to new staff positions that reflect an evolving understanding of technical issues?"

> An evolving understanding of issues does not, by itself, define which category fits a particular backfit. Judgment must be applied to the facts of each particular case to determine whether the backfit is for compliance, to provide adequate protection, to redefine adequate protection, or to achieve a cost-justified substantial safety enhancement. For example, with regard to compliance, the 1985 statement of considerations for 10 CFR 50.109 indicates that "the compliance exception is intended to address situations where the licensee has failed to meet known and established standards of the Commission because of omission or mistake of fact....new or modified interpretations of what constitutes compliance would not fall within the exception...."

NUREG-1409 also provides an example where an evolving understanding of technical issues resulted in a compliance backfit that was apparently justified for at least some licensees. In

response to industry claims that Bulletin 88-11[71] lacked any backfitting justification, the NRC staff responded:

> Although the justification was not printed in the bulletin, NRC Bulletin 88-11, "Pressurizer Surge Line Thermal Stratification," was justified as a backfit. It is an example of a backfit that was determined by the responsible NRC official to be required as a matter of compliance with existing requirements and commitments. The CRGR reviewed the bulletin and concurred. The regulations currently require licensees to meet the applicable codes of the American Society of Mechanical Engineers (ASME), *Boiler and Pressure Vessel Code*. Because of the NRC staff's concern with the integrity of the surge line, licensees were requested to perform their fatigue analysis in accordance with the latest ASME Section III requirements that incorporate high cycle fatigue analysis. The justification provided by the NRC staff was that previously unconsidered thermal stratification phenomenon may invalidate the existing analysis performed to confirm the integrity of the surge line.
>
> Subsequently, it was understood that some licensees believed that the NRC staff's rationale was in error because they were not committed to the latest ASME Section III requirements by virtue of their license commitment. However, the issue became moot because these licensees undertook the analysis voluntarily in view of the safety importance of the issue and the fact that previous versions of the ASME Code did not completely address the concern.

[71] NRC 1988e

APPENDIX B: QUALIFICATION OF PRESSURE RELIEF VALVES IN NUCLEAR POWER PLANTS IN RESPONSE TO THE TMI-2 ACCIDENT

Nuclear power plants in the United States use various types of pressure relief valves to protect personnel and equipment from overpressure events within reactor fluid systems. Pressure relief valves include safety valves, safety relief valves, and relief valves, with different designs, operating conditions, and requirements. The American Society of Mechanical Engineers (ASME) *Boiler and Pressure Vessel Code* (BPV Code), Section III, Division 1, specifies requirements for the design, operation, installation, and testing of pressure relief valves used for various functions in nuclear power plants.[72] For example, the ASME BPV Code (2007 Edition) in Article NB-7000, Overpressure Protection, specifies requirements several service conditions:

- steam and air or gas service for safety valves;
- steam, air or gas, and liquid service for safety relief valves;
- liquid service for relief valves; and
- steam, air or gas, and liquid service for pilot operated or power actuated pressure relief valves.

The ASME *Code for Operation and Maintenance of Nuclear Power Plants* (OM Code) provides requirements for the preservice and inservice testing (IST) programs for pressure relief valves in nuclear power plants.

Braidwood, Units 1 and 2 (Braidwood) and Byron, Units 1 and 2 (Byron) are Westinghouse-designed pressurized-water reactors (PWRs) that received their construction permits under Title 10 of the *Code of Federal Regulations* (10 CFR), Part 50, in December 1975. The pressurizer for each unit is equipped with three pressurizer safety valves (PSVs) and two power-operated relief valves (PORVs). The three PSVs are Crosby Model HP-BP-86, size 6M6 (6-inch), spring loaded pop type opened by direct fluid pressure. The PORVs are Copes-Vulcan Model D-100-160 3-inch pneumatic-actuated globe valves that respond to a signal from the pressure sensing system or to manual control. Each PORV can be isolated by a motor-operated block valve.

The ASME BPV Code of record for the PSVs at Byron and Braidwood was the 1971 Edition through the Winter 1972 addenda of the ASME BPV Code, Section III. The ASME BPV Code applicable to Byron and Braidwood includes requirements for overpressure protection. For example, Section NB-7300, "Overpressure Protection Report," in NB-7320(f) requires that the report include the redundancy and independence of the pressure-relief devices and their associated pressure-sensing and controls systems employed to preclude a loss of overpressure protection in the event of a failure of any pressure-relief device, or its sensing element, or its associated control, or an external power source. NB-7411, "Relieving Capacity of Pressure-Relief Devices," specifies that the total rated relieving capacity shall be sufficient to prevent a rise in pressure of more than 10% above system design pressure (at design temperature) within the pressure-retaining boundary of the system under any pressure transient anticipated to arise as summarized in the Overpressure Protection Report. NB-7421, "Required Number and

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[72] References to individual ASME Code publications are not provided in Appendix D, but they are publicly available from ASME for a fee.

Capacity of Pressure-Relief Devices for Nuclear Systems," states that the required relieving capacity intended for overpressure protection of a nuclear power system or portions of the system shall be secured by the use of at least two pressure-relief devices.

At the time of the Byron and Braidwood operating license review, NRC Standard Review Plan (SRP), Revision 1 (July 1981), Chapter 15.5.1-15.5.2, "Inadvertent Operation of ECCS and Chemical and Volume Control System Malfunction that Increases Reactor Coolant Inventory," and Chapter 15.6.1, "Inadvertent Opening of a PWR Pressurizer Pressure Relief Valve or a BWR [boiling-water reactor] Pressure Relief Valve," provided general staff guidance for these plant transients. In March 2007, the NRC staff issued Revision 2 to these SRP sections with significantly more detail, including a statement that PSVs and PORVs are assumed to fail open if they relieve water without being qualified.

The accident at Three Mile Island, Unit 2 (TMI-2) on March 28, 1979, included failure of a PORV on the pressurizer to reclose properly during the event. Based on lessons learned from the TMI-2 accident, the NRC issued recommendations regarding performance testing of safety and relief valves used in nuclear power plants in NUREG-0578 (July 1979), "TMI-2 Lessons Learned Task Force Status Report and Short-Term Recommendations." In particular, the NRC staff recommended in Section 2.1.2, "Performance Testing for BWR and PWR Relief and Safety Valves," of NUREG-0578 that nuclear power plant licensees commit to provide performance verification by full-scale prototypical testing for all relief and safety valves.

On October 31, 1980, the NRC issued a letter to all then-operating nuclear power plants and applicants for operating licenses and holders of construction permits forwarding NUREG-0737, "Clarification of TMI Action Plan Requirements." Requirement II.D.1, "Performance Testing of Boiling-Water Reactor and Pressurized-Water Reactor Relief and Safety Valves (NUREG-0578, Section 2.1.2)," in NUREG-0737 specified the NRC position that PWR and BWR licensees and applicants shall conduct testing to "qualify" the reactor coolant system (RCS) relief and safety valves under expected operating conditions for design-basis transients and accidents. The detailed clarification in NUREG-0737 of this NRC position specified the following:

> Licensees and applicants shall determine the expected valve operating conditions through the use of analyses of accidents and anticipated operational occurrences referenced in Regulatory Guide 1.70, Revision 2. The single failures applied to these analyses shall be chosen so that the dynamic forces on the safety and relief valves are maximized. Test-pressures shall be the highest predicted by conventional safety analysis procedures. [RCS] relief and safety valve qualification shall include qualification of associated control circuitry, piping, and supports, as well as the valves themselves.
>
> A. Performance Testing of Relief and Safety Valves--The following information must be provided in report form by October 1, 1981:
>
> (1) Evidence supported by test of safety and relief valve functionability for expected operating and accident (non-ATWS) conditions must be provided to NRC. The testing should demonstrate that the valves will open and reclose under the expected flow conditions.
>
> (2) Since it is not planned to test all valves on all plants, each licensee must submit to NRC a correlation or other evidence to substantiate that the valves

tested in the EPRI (Electric Power Research Institute) or other generic test program demonstrate the functionability of as-installed primary relief and safety valves. This correlation must show that the test conditions used are equivalent to expected operating and accident conditions as prescribed in the final safety analysis report (FSAR). The effect of as-built relief and safety valve discharge piping on valve operability must also be accounted for, if it is different from the generic test loop piping.

(3) Test data including criteria for success and failure of valves tested must be provided for NRC staff review and evaluation. These test data should include data that would permit plant-specific evaluation of discharge piping and supports that are not directly tested.

In describing the type of review to be conducted for this regulatory position, the NRC staff stated the following:

Pre-implementation review will be performed for EPRI and BWR test programs with respect to qualification of relief and safety valves. Also, the applicants' proposal for functional testing or qualification of PWR valves will be reviewed. Post-implementation review will also be performed of the test data and test results as applied to plant-specific situations.

In specifying the documentation required to satisfy this regulatory position, the NRC staff stated the following:

Pre-implementation review will be based on EPRI, BWR, and applicant submittals with regard to the various test programs. These submittals should be made on a timely basis as noted below, to allow for adequate review and to ensure that the following valve qualification dates can be met:

Final PWR (EPRI) Test Program--July 1, 1980

Final BWR Test Program--October 1, 1980

Block Valve Qualification Program--January 1, 1981

Post-implementation review will be based on the applicants' plant-specific submittals for qualification of safety relief valves and block valves. To properly evaluate these plant-specific applications, the test data and results of the various programs will also be required by the following dates:

PWR (EPRI)/BWR Generic Test Program Results--July 1, 1981

Plant-specific submittals confirming adequacy of safety and relief valves based on licensee/applicant preliminary review of generic test program results--July 1, 1981

Plant-specific reports for safety and relief valve qualification--October 1, 1981

Plant-specific submittals for piping and support evaluations--January 1, 1982

Plant-specific submittals for block valve qualification--July 1, 1982.

In a letter dated July 27, 1982, to the NRC staff, the Westinghouse Owners Group (WOG) submitted WCAP-10105 (June 1982), "Review of Pressurizer Safety Valve Performance as Observed in the EPRI Safety and Relief Valve Test Program." In WCAP-10105, the WOG indicated that the design specification for PSVs in Westinghouse-designed nuclear power plants is for steam service only. Based on a review of the EPRI test data, the WOG concluded that the valves performed with chatter, but did not identify any valve damage. (ADAMS LL Accession No. 8208190310, Microfiche 14387:191-301)

In December 1982, EPRI issued NP-2628-SR, "EPRI PWR Safety and Relief Valve Test Program – Safety and Relief Valve Test Report," that described safety and relief valve tests for types of valves in service at nuclear power plants. In particular, Section 3.5 in EPRI NP-2628-NP discusses the testing of Crosby safety valves similar to the PSVs at Byron and Braidwood, including two water tests. The report indicated chattering of the safety valves with subsequent inspection finding galled surfaces and damage to internal parts. Section 4.6 in EPRI NP-2628 discussed testing of Copes-Vulcan relief valves similar to the pressurizer PORVs at Byron and Braidwood, although the extent of water testing is not fully described. The report indicated no damage found during the inspection of the Copes-Vulcan relief valves. The report did not indicate any failures of the Crosby or Copes-Vulcan valves to reseat during the testing. (ADAMS LL Accession No. 8407130197, Microfiche 25588:082-262)

In January 1983, EPRI issued NP-2770-LD, "EPRI/C-E PWR Safety Valve Test Report," that described the testing of PWR primary system safety valves. Volume 1 provides a summary of the test program and its results. Section 4.5 of Volume 1 indicates that the following tests were performed on the Crosby 6M6 PSV: 11 steam tests with filled loop seals, 3 steam-to-water transition tests, and 2 water tests. The report states that the valve experienced chatter during the tests, and one water test had to be terminated. The individual volumes of EPRI NP-2770-LD discuss the test results for each specific PSV type. Volume 6 provides the test details for the Crosby 6M6 PSV. (EPRI NP-2770-LD, Volume 1, was obtained as a public document from the EPRI website. EPRI NP-2770-LD, Volume 6, could not be located within ADAMS or the NRC Record Retention Files, but is available for a fee from EPRI.)

In October 1982, EPRI issued NP-2670-LD, "EPRI/Wyle Power-Operated Relief Valve Phase III Test Report," to address testing of PORVs. This document could not be located in ADAMS despite its reference by nuclear power plant licensees. See, for example, North Anna Units 1 and 2 UFSAR (Revision 51, dated September 30, 2015), Section 15.2.14, "Spurious Operation of the Safety Injection System at Power."

The NRC review of the operating license applications for Byron and Braidwood included evaluation of the TMI Action Plan items as discussed in the NRC Safety Evaluation Report (SER) for Braidwood Units 1 and 2, NUREG-1002, Section 1.1, "Introduction." In this SER section, the NRC staff stated that the review and evaluation of compliance by the applicant with the licensing requirements established in NUREG-0660, "NRC Action Plan Developed as a Result of the TMI-2 Accident," and TMI Action Plan Item II.D.1 were incorporated into the reviews summarized throughout the SER. The NRC SER for Byron Units 1 and 2, NUREG-0876, also includes discussions of the NRC staff review of the TMI Action Plan items.

Appendix E, "Requirements Resulting from TMI-2 Accident," to the Byron and Braidwood UFSAR in Section E.23, "Relief and Safety Valve Test Requirements (II.D.1)," indicated that a letter dated April 1, 1982, from D. Hoffman (Consumers Power) transmitted the Safety and Relief Valve Test Report for the EPRI PWR Safety and Relief Valve Test Program. The UFSAR stated that the final evaluation of the data indicated that the relief and safety valves will perform their intended functions for all expected fluid inlet conditions. The UFSAR also indicated that the plant-specific final evaluation confirming the adequacy of the relief and safety valves had been submitted by a letter from T. Tramm, dated October 26, 1982.

In Supplement 1 to the Braidwood SER (NUREG-1002, Supplement 1, September 1986), in Section 3.9.3.3, "Design and Installation of Pressure Relief Devices," the NRC staff stated that EPRI had completed a full-scale valve testing program, and that the WOG submitted the test results in WCAP-10105 in a letter dated July 27, 1982, from O. Kinglsey to S. Chilk. (ADAMS LL Accession No. 8208190307, Microfiche 14387:189-301) The NRC staff stated that the applicant responded to a requirement to demonstrate operability of these valves through submittals dated July 1 and October 26, 1982, and December 30, 1983. On the basis of a preliminary review, the NRC staff concluded that the applicant's general approach to responding to this item was acceptable, and provided adequate assurance that the RCS overpressure protection systems at Braidwood can adequately perform their intended functions. The NRC staff stated that if the detailed review revealed modifications or adjustments to safety valves, PORVs, PORV block valves, or associated piping, were needed to ensure that all intended design margins were present, the NRC staff would require that the applicant make appropriate modifications. The NRC staff categorized this issue as a Confirmatory Item. In Supplement 5 to the Byron SER (NUREG-0876, Supplement 5, October 1984) in Section 3.9.3.3, the NRC staff provided a similar discussion of the status of the NRC review of the capability of the Byron pressurizer valves. In Supplement 8 to the Byron SER (March 1987), the NRC staff stated TMI Action Plan Item II.D.I (3.9.3.3) had been closed in Supplement 5 to the Byron SER. The NRC issued operating licenses for Byron Unit 1 in February 1985 and Unit 2 in January 1987, and Braidwood Unit 1 in July 1987 and Unit 2 in May 1988.

Following the issuance of the Byron and Braidwood operating licenses, the NRC staff provided a letter dated August 18, 1988, from L. Olshan to H. Bliss, indicating that Idaho National Engineering Laboratory (INEL) Technical Evaluation Report (TER) EGG-NTA-8028 (January 1988) provided the review of the Byron response to TMI Action Plan Item II.D.1. (ADAMS LL Accession No. 8808260355, Microfiche 46653:240-269) The NRC staff indicated that the licensee should develop and adopt plant procedures to inspect the pressurizer valves after each lift involving loop seal or water discharge. The TER described the INEL review of the EPRI testing of a PSV and PORV similar to the Byron pressurizer valves. Section 4.2.3, "Extended High Pressure Injection [HPI] Event," of the TER stated that the potential for water discharge in extended HPI events can be disregarded for an extended high pressure injection event because at least 20 minutes would be available for operator action. However, Section 4.2.2, "FSAR Liquid Transients," of the TER discussed the evaluation of the PSVs and PORVs for feedwater line break accidents that would include water discharge, and determined that the EPRI tests were applicable to the Byron/Braidwood PSVs and PORVs. In addition, Section 4.3.1, "Safety Valves," and Section 4.3.2, "Power Operated Relief Valves," of the TER determined that the performance of the PSVs and PORVs was acceptable based on the EPRI tests, including water discharge tests. The TER indicated that the PSV had two applicable tests: a loop seal/steam water transition test where the valve opened, chattered and stabilized to close; and a saturated water test where the valve opened with water, chattered, and stabilized. The TER indicated that the PORV opened and closed on demand in the loop seal/steam water transition test with a bending moment that was evaluated by analysis. The TER concluded that Byron provided an acceptable response to TMI Action Plan Item II.D.1. On May 21, 1990, the NRC staff provided a letter from S. Sands to T. Kovach with the Braidwood TER that included similar findings. (ADAMS LL Accession No. 9005290209, Microfiche 53927:301-330)

In January 1988, WCAP-11677, "Pressurizer Safety Relief Valve for Water Discharge During a Feedwater Line Break," provided a description of the WOG comparison of the EPRI test data with feedwater line break safety analyses. This report was submitted as an attachment to a response to a request for additional information (RAI) dated May 8, 1989, from the licensee of the Seabrook nuclear power plant. (ADAMS Microfiche 49775:336 – 49756:017) As discussed in the report, the WOG determined that all nuclear power plants addressed in the EPRI testing have PSVs that will operate reliably during water discharge. The WOG evaluated the performance of the Crosby 6M6 PSVs during the EPRI tests, and considered that the performance involved less significant flutter (half lift motion) than the chatter (full lift motion) determined in the EPRI report. The WOG concluded that the Crosby 6M6 PSV can pass slightly subcooled water at a minimum up to three times without damage.

APPENDIX C: CONCERNS REGARDING PERFORMANCE OF PRESSURIZER VALVES UNDER WATER FLOW CONDITIONS

Westinghouse Nuclear Safety Advisory Letter

In 1993 and 1994, Westinghouse issued Nuclear Safety Advisory Letter (NSAL) 93-013 (June 30, 1993) and NSAL-93-013, Supplement 1 (October 28, 1994) to operating nuclear power plants (including Byron and Braidwood). These advisories resulted from Westinghouse's discovery that potentially nonconservative assumptions were used in the licensing analysis of the Inadvertent Operation of the Emergency Core Cooling System at Power (IOECCS) event.

In NSAL-93-013, Westinghouse recommended that licensees determine if their pressurizer safety relief valves (PSRVs) are capable of closing following discharge of subcooled water. Westinghouse noted that the PSRVs might have been designed or "qualified" to relieve subcooled water. Westinghouse indicated that water discharge through the power-operated relief valves (PORVs) is not a concern, because the PORV block valves can be used to isolate the PORVs if they fail to close. If the PSRVs are not designed or qualified for subcooled water discharge, Westinghouse recommended that licensees re-evaluate the IOECCS event with three possible options of (1) reducing ECCS flow used in the safety analysis, (2) using a less restrictive operator response time, or (3) crediting the use of one or more PORVs to help mitigate the event.

In Supplement 1 to NSAL-93-013, Westinghouse informed licensees of a potential reduced time for operator action if a positive displacement pump is in service, and to the need to qualify the PSRVs and the piping downstream of the PSRVs and PORVs if water discharge from the pressurizer is predicted.

Some licensees of operating nuclear power plants informed the NRC of their actions to address the potential concerns regarding liquid service for pressurizer safety valves (PSVs) and PORVs. A sample of actions by nuclear power plant licensees is summarized below in the "Plant-Specific Actions" section.

Additional NRC Generic Communications and Guidance

In December 2003, the NRC staff issued NRR Review Standard for Extended Power Uprates (RS-001, Rev. 0). Item 8 on page 7 of the review standard states that pressurizer level should not be allowed to reach a pressurizer water-solid condition.

On December 14, 2005, the NRC issued Regulatory Issue Summary (RIS) 2005-29, "Anticipated Transients that could Develop into More Serious Events," to notify nuclear power plant licensees of a concern identified during recent reviews of power uprate LARs. In RIS 2005-29, the NRC staff stated that typically ANS Condition II event scenarios involve discharging water through relief or safety valves that are not qualified for water discharge. The NRC staff stated that these valves are then assumed to fail in the open position and create a small break LOCA. The NRC staff stated that it was concerned that some licensees may be crediting PORVs without qualification for water discharge and without establishing additional restrictions to ensure the availability of PORVs and block valves. The NRC staff stated that Westinghouse NSAL-93-013 allowing block valves to isolate PORVs is inconsistent with non-escalation position.

In proposed Revision 1 to RIS 2005-29, the NRC staff addresses the specific ANS Condition II scenarios of chemical volume and control system (CVCS) malfunction, IOECCS event, and inadvertent opening of a PORV or PSV. Regarding the CVCS malfunction, the NRC staff states that performing only the reactivity anomaly analysis or assuming that this malfunction is not as severe as the IOECCS event is not acceptable. Regarding the IOECCS event, the NRC staff states that five of the alternative approaches in NSAL-93-013 fail to meet the non-escalation position. The NRC staff indicated that these unacceptable alternative approaches are (1) closing the block valve, (2) assuming that the PORV is not operable, (3) addressing a stuck-open PORV or PSV as a separate ANS Condition II event, (4) determining that a stuck-open PORV or PSV is not as severe as a small break LOCA, and (5) determining that RCS loss through PORV is made up by ECCS flow. Regarding inadvertent opening of PORV or PSV, the NRC staff states that inadvertent opening of PSV or PORV could continue as an ANS Condition III small break LOCA and fails to meet the non-escalation position.

Additional General PSV/PORV Information

In August 2004, EPRI issued Report 1011047, "Probability of Safety Valve Failure-to-Reseat Following Steam and Liquid Relief - Quantitative Expert Elicitation," 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.

In March 2011, the NRC published NUREG/CR-7037, "Industry Performance of Relief Valves at U.S. Commercial Nuclear Power Plants through 2007," based on a study by the Idaho National Laboratory. With respect to pressurizer PORVs, the report found four separate liquid relief events at four PWR plants. The report estimated 698 total demands on these PORVs during their liquid relief events with no failures to close. The report also summarized test data from EPIX for three valve types. The report indicates 2 failures of PORVs to reclose during 2070 demands, but does not specify liquid or steam service for the EPIX test information. With respect to PSVs, the report indicates 2 failures out of 4 total demands following plant scrams, but does not indicate liquid or steam service. NRC staff from the Office of Nuclear Regulatory Research provided Licensee Event Report information indicating that the 2 PSV failures involved reseating of the valves with leakage of 25 and 200 gallons per minute, respectively. The report summarized EPIX test data for PSVs as no failures to reclose during 1805 demands.

Plant-Specific Actions

Diablo Canyon

On August 13, 1996, the licensee of the Diablo Canyon nuclear power plant submitted a report under 10 CFR 50.59 related to the potential for an IOECCS event. (ADAMS Microfiche 89419:294-322) The submittal included NSAL-93-013 and its supplement as enclosures. The licensee indicated that the PSVs had not been initially qualified for water discharge, but were subsequently qualified for a brief period. The licensee indicated that WCAP-11677 was applicable and demonstrated that the PSVs were operable.

On July 2, 2004, the NRC granted a license amendment request (LAR) for Diablo Canyon that allowed credit for actuation of the PORVs in response to inadvertent safety injection (SI) actuation to avoid challenges to the PSVs. (ADAMS Accession No. ML041950300) In support of that LAR, the licensee responded on November 21, 2003, to requests for additional information (RAIs) related to the capability of the PORVs to function adequately under conditions predicted for design-basis transients and accidents. (ADAMS Accession No. ML033360735) In response to an RAI regarding the design adequacy of the PORVs if the pressurizer becomes water solid, the licensee had stated that the NRC had issued a letter dated January 26, 1986, "Safety and Relief Valve Testing, NUREG-0737 Item II.D.1," that provided an SER that accepted the adequacy of the PORV and block valve design and confirmatory testing for a range of fluid conditions (full pressure steam, steam to water transition, and subcooled water fluid).

Salem

On June 4, 1997, the NRC granted a technical specification (TS) revision for the Salem nuclear power plant to ensure that the automatic capability of the PORVs to relieve pressure is maintained. (ADAMS Accession No. ML011720397) In response to NSAL-93-013, the licensee determined that an inadvertent SI actuation at power could cause the pressurizer to become water solid and PSVs lifting with water discharge if the automatic operation of the PORVs is not made available for reactor coolant system (RCS) depressurization early in the transient. In that the Salem PSVs were not designed to relieve water, it was noted that water discharge has the potential to cause the PSVs to fail in the open position.

In the course of the review of the licensee's application, the NRC staff noted that the PORVs were not designed to "safety related" standards and, thus, could not be credited for mitigation of the inadvertent SI actuation at power incident when the PORV is operating in the automatic mode. In response, the licensee proposed an upgrade of the PORVs to eliminate the possibility that a single active failure of a PORV component could prevent the mitigation of the inadvertent SI actuation at power incident. As discussed in the SER, the licensee implemented modifications to the PORV circuitry to qualify the upgraded circuitry as safety-related.

Regarding PORV performance, the licensee evaluated the PORV air accumulators for sufficient capacity for the inadvertent SI event. The licensee also reported that endurance tests had been performed with five different trims (with different trim materials) on one PORV at Wyle Laboratories to demonstrate that (1) after 2000 consecutive operations, there were no packing leaks nor packing gland adjustments required; (2) there was no diaphragm failure; and (3) the solenoid valve withstood 10,000 operations without any loss of function. Based on this information, the NRC staff concluded that the PORV performance was acceptable regarding the mitigation of an inadvertent SI event.

Millstone, Unit 3

On June 5, 1998, the NRC granted a license amendment for Millstone, Unit 3 for a TS revision to ensure that the capability of the PORVs to relieve pressure is maintained. (ADAMS Accession No. ML011800207) The revised TS Bases stated that the PORVs and their associated piping have been demonstrated to be "qualified" for water discharge. The PORVs prevent water discharge from the PSVs for which qualification for water discharge has not been demonstrated. The TS Bases also stated that the prime importance for the capability to close the block valve is to isolate a stuck-open PORV. In the SER, the NRC staff stated that the licensee notified the NRC of the issue of potential water discharge through the PSVs that could lead to valve failure in LER 97-063-00 on December 31, 1997.

To provide added assurance that the PSVs will not be damaged due to water discharge during an ISI event, the licensee upgraded the PORV circuitry, added additional PORV surveillance requirements, qualified the PORVs and associated piping for water discharge, and made emergency procedure changes to allow plant operators additional time to terminate the event. With respect to the PORV circuitry, the NRC staff concluded that the PORV circuitry modifications qualified the PORV control circuitry as safety-related. With respect to PORV performance, the licensee reanalyzed the inadvertent SI event with the LOFTRAN computer code to demonstrate that the PORVs were qualified for water discharge for approximately 1 hour. The licensee referenced EPRI testing documented in NP-2670-LD, Volume 11, that was said to generically resolve post TMI-2 issues associated with PORVs and safety valve qualification for water and steam relief, with the results from four tests of a Garrett PORV (such as used at Millstone, Unit 3) for water discharge. The licensee determined that the PORVs and associated piping are qualified for 1 hour of water discharge for an IOECCS event. The licensee also stated that the PORV manufacturer performed numerous cycle tests to verify the performance of the valve design, and also verified that valve seat leakage was acceptable. The licensee stated that the PORV block valves had been evaluated for water discharge in accordance with the program established in response to Generic Letter (GL) 89-10, "Safety-Related Motor-Operated Valve Testing and Surveillance." The NRC staff found the licensee information regarding the qualification of the PORVs for water discharge during the inadvertent SI event to be acceptable.

Callaway

On September 25, 2000, the NRC granted a license amendment for the Callaway nuclear power plant to revise the TS to change the PSV lift setting range. (ADAMS Accession No. ML003753326) To prevent water passing through the PSVs during an IOECCS event, the licensee modified and upgraded the PORV circuitry to full Class 1E to take credit for automatic action of at least one PORV during the event. These actions would prevent water discharge through the PSVs. In its TS revision request dated May 25, 2000, the licensee had stated that the design function of the valves was not being changed and the conclusions documented in the NRC SER of Callaway's response to NUREG-0737 Item II.D.1 (dated September 10, 1987) are unchanged. As a result, the licensee stated that the PORVs and associated discharge piping can accommodate water discharge.

Byron and Braidwood

On May 29, 1998, the licensee for Byron and Braidwood proposed an amendment to its TS to take credit for the automatic operation of the PORV to provide mitigation for an IOECCS event. In the amendment request, the licensee stated that the PSVs have not been qualified to reseat after passing subcooled liquid. The licensee stated that the PORVs at Byron and Braidwood are safety-related components with safety-related actuators and accumulator tanks with PORV control circuits classified as safety-related. The licensee noted that some portions of the PORV circuitry are nonsafety-related with improvements implemented in response to GL 90-06, Resolution of Generic Issue 70, "Power-Operated Relief Valve and Block Valve Reliability" and Generic Issue 94, "Additional Low-Temperature Over Pressure Protection for Light-Water Reactors" Pursuant to 10 CFR 50.54(f)." The licensee stated that the PORV block valves are within the scope of the GL 89-10 program. In a letter dated May 13, 1999, the NRC staff provided an RAI regarding the reliance on the PORVs that documented the basis for its concerns that the PORV circuitry did not meet the single failure criterion. In response to these concerns, the licensee withdrew its TS amendment request in a letter dated July 16, 1999. No further action regarding this amendment request has been identified. However, the licensee stated during a public meeting with the NRC staff on March 7, 2016, to discuss this backfit issue that the PORVs and their block valves at Byron and Braidwood are safety-related with the exception of one circuitry aspect of the PORV (see March 7, 2016, transcript, pages 36-40).

On July 5, 2000, the licensee for Byron and Braidwood submitted a request for a power uprate for Byron and Braidwood to increase the maximum thermal power for each unit from 3411 megawatts thermal (MWt) to 3586.6 MWt (commonly referred as a stretch power uprate). In RAIs, the NRC staff requested that the licensee address water solid conditions in the pressurizer because it had generally not accepted a solid pressurizer for an IOECCS event to order to avoid the potential for all three PSVs to be stuck open due to liquid relief through these safety valves. In its letter dated November 27, 2000, the licensee stated that Section 15.5.1, "Inadvertent Operation of Emergency Core Cooling System During Power Operation," of the UFSAR had been revised to credit the PSVs to pass water. The licensee discussed the EPRI testing program in response to NUREG-0737 with the results summarized in EPRI NP-2628-SR. The licensee referenced the NRC letters from L. Olshan to H. Bliss, dated August 18, 1988, and S. Sands to T. Kovach, dated May 21, 1990, transmitting the TERs with the results of the NRC's review of the Byron and Braidwood response to TMI Action Plan Item II.D.1, respectively.

On January 31, 2001, the licensee for Byron and Braidwood provided a response to an RAI supplement from the NRC staff requesting the temperature of water to be passed by the pressurizer safeties and the length of time that the safeties are expected to pass water. The NRC staff also asked the licensee to discuss what EPRI tests are applicable to the Byron and Braidwood condition. In response, the licensee stated that the PSVs would close after passing water, although they may not be leaktight. The licensee stated that the leakage from up to three leaking PSVs is bounded by one fully open PSV. The licensee indicated that the EPRI testing of the Crosby safety valves in EPRI NP-2770-LD, Volumes 1 and 6, are applicable. The licensee indicated that valve chatter occurred during the tests with damage to the internals, but that the safety valve closed in response to system depressurization. The licensee stated that the Byron/Braidwood pressurizer water temperature of 590 °F is higher than the EPRI tests (530 °F). The licensee stated that the assumed length of the event is 20 minutes from initial SI signal to when the system pressure is restored below PSV lift setpoint.

In the NRC SER dated May 4, 2001, granting the Byron/Braidwood power uprate in Section 3.2, "Non-LOCA [loss-of-coolant accident] Transient Analysis," the NRC staff discussed its review of the performance of the PORVs and PSVs to discharge liquid water for approximately 20 minutes. (ADAMS Accession No. ML033040016) The NRC staff discussed the EPRI testing program with the conclusion that the safety valve closed in response to system depressurization. The NRC staff reviewed the licensee's evaluation of the performance of the PSVs for liquid water conditions. The NRC staff found that the EPRI tests adequately demonstrate the performance of the valves for the expected water temperature conditions and that there is reasonable assurance that the valves will adequately reseat following the spurious SI event. The NRC staff determined that a review of the EPRI test data indicates that the PSVs may chatter for the expected fluid inlet temperature, but that the resulting PSV seat leakage following the water discharge would be less than the discharge from one stuck-open PSV. Therefore, the NRC staff found the licensee's crediting of the PSVs to discharge liquid water during the spurious SI event to be acceptable. This portion of the NRC SER was based on the specific review of PSV performance for the Byron and Braidwood power uprate request described in a memorandum dated March 15, 2001, from the NRR Reactor Systems Branch with technical input from the responsible staff member for safety valves in the NRR Division of Engineering (ADAMS Accession No. ML010740316).

As noted by the licensee, the Byron/Braidwood UFSAR at the time of the stretch power uprate (Revision 9, dated December 2002) in Chapter 15.5.1 includes PSV water discharge, and references the INEL 1988 report and L. Olshan August 1988 SER. The current UFSAR Revision 15 (dated December 2014) concludes that the IOECCS event does not progress into a stuck-open PSV LOCA event. The UFSAR states that all three PSVs may lift but will reclose, and that the leakage is bounded by one fully open valve with the consequences bounded by the IOPSRV event. The UFSAR also specifies that if a safety injection results in discharge of coolant through the pressurizer valves, the operators will bring the plant to cold shutdown to inspect the valves.

On August 26, 2004, the NRC issued a license amendment for Byron and Braidwood granting an adjustment to the PSV setpoints. (ADAMS Accession No. ML042250531) In an RAI, the NRC staff requested that the licensee perform a quantitative analysis regarding PSV water cycles and relief/discharge water temperature. As for the loss of ac power (LOAC) with reactor coolant pump (RCP) seal injection event, the licensee's analysis indicated that continued injection of water into the RCS through the RCP seals would result in a water-solid pressurizer and water discharge through the PSVs. The proposed PSV setpoint tolerance assuming negative tolerance would result in a lower PSV lift setpoint. With the lower setpoint, the PSV would open earlier, and a larger number of PSV water cycles with a lower water discharge temperature could result during the transient. The licensee performed an analysis of the LOAC with RCP seal injection event, and determined the revised PSV setpoint would result in an increase of about one PSV water cycle and a reduction in the water discharge temperature of about 0.5 °F. A comparison of the reanalysis showed that the spurious SI event remained the limiting event since it resulted in a greater increase in the number of PSV water cycles (two cycles vs. one cycle) and a greater decrease in the PSV discharge water temperature (3.0 °F vs. 0.5 °F) than that calculated for the LOAC with RCP seal injection event. The water discharge temperature in the analysis of record for the spurious SI event was 590 °F. The lowest discharge water temperature for the spurious SI event with the revised PSV setpoint is 587 °F. The NRC staff found that the calculated water discharge temperature (587 °F) is significantly higher than the discharge water temperature of 530 °F that was used to support operability of the PSVs as discussed in the analysis of record. As a result, the NRC staff concluded that the reanalysis is acceptable to assure that the PSVs will remain operable following a spurious SI event.

On February 7, 2014, the NRC issued a license amendment for Byron and Braidwood granting a Measurement Uncertainty Recapture (MUR) power uprate. The NRC staff determined that the IOECCS event was outside of the scope of the MUR power uprate, because the licensee did not modify the Chapter 15 analyses related to PSV and PORV water discharge.

With respect to inservice testing (IST) activities, the Byron IST Program (dated July 21, 2016) references the ASME *Code for Operation and Maintenance of Nuclear Power Plants* (OM Code), 2004 Edition through 2006 Addenda; and the Braidwood IST Program (dated July 27, 2009) references the ASME OM Code, 2001 Edition through 2003 Addenda. The Byron IST Program specifies the following testing and intervals for the PORVs, PORV block valves, and PSVs:

- PORV - fail safe test closed (cold shutdown interval); stroke-time exercise open and closed (cold shutdown interval); and position indication test (2 year interval).

- PORV Block Valve- exercise open and closed (2 year interval); position indication test (Joint Owners Group (JOG) Program interval); and open and closed test in accordance with ASME OM Code Case OMN-1, "Alternative Rules for Preservice and Inservice Testing of Active Electric Motor Operated Valve Assemblies in Light-Water Reactor Power Plants" (JOG Program interval).

- PSV – Relief Valve Test (5 year interval); and position indication test (2 year interval). The Byron IST Program references Appendix I, "Inservice Testing of Pressure Relief Devices in Light-Water Reactor Nuclear Power Plants," to the ASME OM Code for Relief Valve Testing.

The Braidwood IST Program specifies the following testing and intervals for the PORVs, PORV block valves, and PSVs:

- PORV - fail safe test closed (refueling outage interval); stroke-time exercise open and closed (refueling outage interval); and position indication test (2 year interval).

- PORV Block Valve- exercise open and closed (quarterly interval); and position indication test (2 year interval).

- PSV – Relief Valve Test (5 year interval); and position indication test (2 year interval). The Braidwood IST Program references Appendix I to the ASME OM Code for Relief Valve Testing.

Shearon Harris

On October 12, 2001, the NRC granted a license amendment to the Shearon Harris nuclear power plant for steam generator replacement and a power uprate to a maximum power level of 2900 MWt (approximately 4.5 percent). In addressing the licensee's evaluation of SRP Section 15.5.1, the NRC staff found that the analysis showed that the calculated inlet pressures and temperatures required for the PORVs and SRVs to operate in a water environment are within the valve operable ranges, and thus ensure that the PORV and SRV are operable during the transient. The valve operable ranges were previously determined by the licensee to support operability of the PORV and SRV during the discharge of subcooled water in accordance with the TMI Action Plan Item II.D.1 requirements. Based on the analysis meeting the acceptance criteria of SRP Section 15.5.1 with respect to the RCS pressure limit and departure-from-nucleate-boiling limit, the NRC staff concluded that the analysis was acceptable.

Beaver Valley

On July 19, 2006, the NRC granted an EPU to Beaver Valley Units 1 and 2 (BVPS-1 and 2) for an approximate 8-percent increase in thermal power to 2,900 MWt. In its SER (ADAMS No. ML061720376), the NRC staff stated that a specific issue which was reviewed related to the capability of the PSVs to discharge liquid and adequately reseat for a spurious SI actuation. The specific issue which the NRC staff evaluated in this regard was whether the PSVs could reasonably be expected to reseat in order to prevent the spurious SI actuation (an ANS Condition II event) from causing a stuck-open PSV (an ANS Condition III event). This issue was said to be further discussed in RIS 2005-29. While the PSVs are qualified to discharge steam, if the valves discharge liquid having a temperature low enough, they may not reseat properly.

Based the licensee's analysis, during a spurious SI event, the PSVs would be required to discharge steam followed by high temperature liquid after the pressurizer fills. The licensee provided plots of the pressurizer water temperatures for this event which indicated that the minimum temperature of the discharged liquid for both BVPS-1 and 2 is approximately 620 °F. To evaluate the capability of the valves to discharge and reseat, the NRC staff reviewed the available data from the full flow tests performed during the EPRI test program in 1981 for the specific PSV models representative of those installed at BVPS-1 and 2. The licensee also used the methodology contained in WCAP-11677, and determined that the minimum acceptable liquid temperature for which the PSVs are expected to successfully discharge and reseat is less than the minimum expected temperature for the spurious SI event for BVPS-1 and 2.

The NRC staff agreed that both the minimum expected water discharge temperature and the minimum acceptable liquid temperature had been conservatively calculated. Therefore, the NRC staff determined that, for purposes of preventing the occurrence of a more serious ANS Condition III event, there is reasonable assurance that the PSVs would adequately discharge and reseat following a spurious SI actuation. A consideration in making this finding was that, in the unlikely event of a stuck-open PSV, the ECCS is fully capable of mitigating the resulting LOCA.

Turkey Point

On June 15, 2012, the NRC granted an EPU for Turkey Point, Units 3 and 4 that increased the thermal power level of each unit approximately 15 percent to 2644 MWt.

In the SER (ADAMS Accession No. ML11293A359), the NRC staff indicated that ECCS actuation is not a possible initiator of inadvertent increase in reactor coolant inventory because the high head SI pumps have a shut-off head below the normal RCS operating pressure. The NRC staff stated that a CVCS malfunction that increases RCS inventory was evaluated for the effects of adding water inventory to the RCS. If the pressurizer fills and causes water to be relieved through the PORVs or safety valves, then these valves could stick open and create a small break LOCA. The NRC staff stated that this would violate the acceptance criterion that prohibits the escalation of an anticipated operational occurrence (AOO) into a more serious event. Satisfaction of this acceptance criterion is demonstrated by showing that sufficient time exists for the operator to recognize the situation and end the charging flow before the pressurizer can fill. The NRC staff concluded that the licensee's analyses of IOECCS and CVCS events adequately accounted for operation of the plant at the proposed power level.

Regarding an inadvertent opening of a pressurizer relief valve, the licensee initially proposed that the consequences of this event are bounded by the small break LOCA. The NRC staff did not accept this proposed disposition. If action is not taken to secure the open valve by either closing the PORV or its block valve, the NRC staff stated that this event could escalate to a small break LOCA, which is contrary to the non-escalation criterion. When the pressurizer becomes water solid, water begins to flow through the open PORV. If the PORV is not qualified for water discharge, the NRC staff stated that it is likely the PORV will not close upon demand. In this way, the NRC staff stated that the inadvertent opening of a PORV, an AOO, becomes a small break LOCA at the top of the pressurizer, an ANS Condition III event. The NRC staff requested that the licensee address the inadvertent opening of the PORV with respect to the third criterion for an ANS Condition II event.

The licensee provided an analysis, performed largely in accordance with NRC-approved, Westinghouse analytic methodology using the RETRAN computer code; however, this analysis was performed assuming that the PORV opened instead of the PSV. The NRC staff stated that assuming the opening of the PORV is acceptable, because the PSV is differently qualified, and reseats mechanically. An additional independent fault would be required to cause the safety valve to fail to close. The analysis indicated that the pressurizer would fill within about 240 seconds. The licensee stated that there are multiple alarms to indicate the opening of a PORV. The licensee stated that a prompt operator action is required to close the PORV and, if the PORV does not close, the operator is to close the block valve. Because the necessary actions are prompt and simple, the NRC staff agreed that there is sufficient time to secure the inadvertently open PORV without filling the pressurizer.

St. Lucie

On September 24, 2012, the NRC granted an EPU for St. Lucie, Unit 2 that increased the authorized thermal power level about 12 percent to 3020 MWt. Regarding an IOECCS event, the high pressure SI pumps are incapable during power operations of delivering flow to the RCS because the pumps' shut-off head is less than the normal RCS operating pressure of 2250 pounds per square inch absolute. Therefore, the inadvertent operation of the ECCS at power event is not a credible event and was not analyzed by the licensee for the proposed EPU. The NRC staff found that the licensee's position for not analyzing the IOECCS event to be acceptable.

Regarding a CVCS malfunction, this event increases RCS inventory as an AOO that is evaluated for the effects of adding water inventory to the RCS. The NRC staff reviewed the licensee's analyses of the CVCS malfunction event and concluded that the licensee's analyses adequately accounted for operation of the plant at the proposed power level and were performed using acceptable analytical models. The NRC staff determined that the licensee's analysis demonstrated that the pressurizer did not become water solid, assuring no water was discharged through the PSVs.

Regarding an IOPORV event, the NRC staff stated that when viewed from the mass addition perspective, this event can be evaluated in two phases: (1) an inadvertent opening of a pressurizer relief valve, followed by (2) an inadvertent ECCS actuation. In the first phase, the NRC staff stated that this event could be mitigated by closing the open pressurizer relief valve or its block valve. If the PORV or its block valve was not closed, the NRC staff stated that the IOPORV event would enter the second phase with actuation of the ECCS. Based on its review, the NRC staff determined that the pressurizer overfill analysis, available alarming system, and procedures in combination with simulator exercise result had provided reasonable assurance that the pressurizer would not be expected to fill to a water solid condition that could prevent the PORV or PSV from closing after they were open, and thus, supported that the event would not generate a more serious plant conditions, meeting the non-escalation criterion. The NRC staff stated that it reviewed the licensee's analyses of the inadvertent opening of a pressurizer pressure relief valve event, and concluded that the licensee's analyses adequately accounted for operation of the plant at the proposed power level and were performed using acceptable analytical models.

The NRC staff concluded that the licensee demonstrated that the all AOO acceptance criteria are satisfactorily met.

North Anna

In UFSAR (Revision 51, dated September 30, 2015) Section 15.2.14, "Spurious Operation of the Safety Injection System at Power," the licensee for North Anna Units 1 and 2 discusses the plant response to an inadvertent SI event. In particular, UFSAR Section 15.2.14.2.3, "Event Propagation," states the following:

> Safety valve (Reference 18) and PORV (Reference 19) testing has revealed no instances of failure of the valves to reseat following water relief. Resulting leakage is within the capacity of the normal makeup system and is therefore not considered to be a small break loss of reactor coolant event. Therefore, the complete filling of the pressurizer and/or water relief via a safety valve as a result of a spurious safety injection does not constitute a failure to meet the event propagation acceptance criterion. Although primary credit for preventing the

propagation of the event to a small break loss of reactor coolant event is the reseating of the PORVs and safety valves, it is noted that the PORVs (which open prior to the safety valves and, if open, preclude safety valve actuation for this event) are provided with block valves which the operator will close in the event of excessive PORV leakage.

North Anna UFSAR Section 15.2.14.3, "Conclusions," states that the complete filling of the pressurizer and/or water discharge via a safety valve as a result of a spurious safety injection does not constitute a failure to meet the event propagation acceptance criterion. In UFSAR Section 15.2, "References," lists Reference 18 as EPRI NP-2770-LD, Volumes 3 and 4, "EPRI/CE PWR Safety Valve Test Reports for Dresser Safety Valve Models 31739A and 31709NA," February and March 1983; and Reference 19 as EPRI NP-2670-LD, Volume 6, "EPRI/Wyle Power-Operated Relief Valve Phase III Test Report, October 1982.

Conclusion

In conclusion, the reliance by the licensee for Byron and Braidwood on the acceptable performance of the PSVs and PORVs for liquid service in response to abnormal events is not inconsistent with similar approaches by some other nuclear power plant licensees. In general, the review of activities by various nuclear power plant licensees related to PSV and PORV performance revealed reliance on EPRI, Wyle, and valve vendor testing to provide support for the performance of these valves under various service conditions. Specific certification for flow capacity of these valves for liquid service in accordance with the ASME BPV Code and National Board was not identified in the review of various justifications prepared by nuclear power plant licensees.

In evaluating the historical documents for Byron and Braidwood, the Panel found it challenging to determine specifically how the licensee resolved the concern raised in NSAL-93-013 in its analyses and plant operations. While the record does support a compliance backfit in this case. If (as recommended by the Panel) the NRC staff undertakes a generic review of licensees' treatment of the potential for pressurizer valve damage following water discharge, it may be appropriate to consider what actions have been taken, how operating experience with water discharge has been considered, and how analysis assumptions are considered in operational practices (including inservice testing) at each plant.

Comment [SW]: Should "does" be "does not?"

Comment [SW]: If "does" is really "does not," change "in this case" to "at this time" ? The would acknowledge the possibility that a generic review could result in regulatory action.

Comment [SW]: I didn't find such a recommendation in the report but I would like for us to make one. Hence, my proposed change at the end of the report.

APPENDIX D: REFERENCES

1. **AEC 1970:** Atomic Energy Commission (AEC), "Backfitting of Production and Utilization Facilities; Construction Permits and Operating Licenses," Title 10 of the *Code of Federal Regulations* (10 CFR), Section 50.109, published March 31, 1970 (as amended). Volume 35 of the *Federal Register* (FR), page 5317.

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3. **ANS 1973:** American Nuclear Society (ANS), ANS-51.1/N18.2-1973, "Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants," dated August 6, 1973. [Publicly available from ANS for a fee.]

4. **ANS 1981:** ANS and American National Standards Institute (ANSI), ANSI/ANS 58.9, "Single Failure Criteria for Light Water Reactor Safety-Related Fluid Systems," issued February 17, 1981 (updated in 2002 and reaffirmed in 2015). [Publicly available from ANS for a fee.]

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9. **ComEd 2000a:** Commonwealth Edison Company, letter from R.M. Krich to U.S. NRC, "Request for a License Amendment to Permit Uprated Power Operations at Byron and Braidwood Stations," dated July 5, 2000. ADAMS Accession No. ML003730608.

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12. **EPRI 1982a:** Electric Power Research Institute (EPRI), NP-2770-LD, "EPRI/C-E PWR Safety Valve Test Report," dated January 1982. [Publicly available from EPRI for a fee.]

13. **EPRI 1982b:** EPRI, NP-2670-LD, "EPRI/Wyle Power-Operated Relief Valve Phase III Test Report," dated October 1982. [Publicly available from EPRI for a fee.]

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18. **Exelon 2014:** Exelon Generation Company, LLC, "Byron/Braidwood Nuclear Stations Updated Final Safety Analysis Report (UFSAR)," Revision 15, dated December 2014. ADAMS Accession No. ML14363A393.

19. **Exelon 2015:** Exelon Generation Company, LLC, letter from J. Bradley Fewell to William M. Dean, U.S. NRC, "Appeal of Imposition of Backfit Regarding Compliance with Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.34(b), General Design Criteria (GDC) 15, GDC 21, GDC 29, and Licensing Basis," dated December 8, 2015. ADAMS Accession No. ML15342A112.

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21. **IAEA 2009:** International Atomic Energy Agency (IAEA), IAEA-TECDOC-1624, "Passive Safety Systems and Natural Circulation in Water Cooled Nuclear Power Plants," dated November 2009. Publicly available from IAEA.

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41. **NRC 1988d:** U.S. NRC, NRC Manual Chapter 0514, "NRC Program for Management of Plant-Specific Backfitting of Nuclear Power Plants," dated August 26, 1988. ADAMS Accession No. ML041400111.

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54. **NRC 2000:** U.S. NRC, letter from Jack Donohew to Garry L. Randolph, Union Electric Company, "Callaway Plant, Unit 1 – Issuance of Amendment Re: Pressurizer Safety Valves and Power Operated Relief Valves (PORVs) (TAC No. MA9080)," dated September 26, 2000. ADAMS Accession No. ML003753326.

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From: West, Steven
To: Clark, Theresa
Cc: Holahan, Gary; Scarbrough, Thomas; Spencer, Michael; West, Steven
Subject: Panel cover letter - suggested edits
Date: Wednesday, August 17, 2016 10:12:48 AM
Attachments: cover memo (MASTER) WEST 2016 08 17.docx

August XX, 2016

MEMORANDUM TO: Victor M. McCree
Executive Director for Operations

FROM: Gary M. Holahan, Backfit Appeal Review Panel Chairman
Office of the Executive Director for Operations

K. Steven West, Deputy Director
Office of Nuclear Security and Incident Response

Thomas G. Scarbrough, Senior Mechanical Engineer
Office of New Reactors

Michael A. Spencer, Senior Attorney
Office of the General Counsel

Theresa Valentine Clark, Executive Technical Assistant
Office of the Executive Director for Operations

SUBJECT: BACKFIT APPEAL REVIEW PANEL FINDINGS ASSOCIATED WITH BYRON AND BRAIDWOOD COMPLIANCE WITH 10 CFR 50.34(b), GDC 15, GDC 21, GDC 29, AND THE LICENSING BASIS

In response to your memorandum of June 22, 2016, establishing a Backfit Appeal Review Panel, the P~~p~~anel undertook a review of the relevant documents in this case. This included the licensee and NRC staff letters; the 2001 power uprate and the 2004 valve setpoint license amendment; and a June 16, 2016, letter from the Nuclear Energy Institute (NEI) supporting the Exelon backfit appeal. The P~~p~~anel also reviewed numerous other documents related to the topic of inadvertent operation of the emergency core cooling system (ECCS) and pressurizer safety valve performance.

In addition to the document review, the P~~p~~anel had the benefit of meetings with the Office of Nuclear Reactor Regulation (NRR) (both the Division of Safety Systems and the Division of Engineering), the Office of the General Counsel (OGC), and the NRC Committee to Review Generic Requirements (CRGR). The Panel also shared its draft preliminary findings with NRR and OGC for comment. NRR provided comments, the consideration of which is reflected in the attached report. Both Exelon (Bradley Fewell) and NEI (Tony Pietrangelo) declined offers for a public meeting but indicated a willingness to provide information if the P~~p~~anel identified the need. The Panel did not identify a need for additional information from either Exelon or NEI to complete its review, which is summarized below and documented in the attached report.

CONTACT: Gary M. Holahan, OEDO

Comment [MAS]: This is the title in the charter.

Comment [SW]: Add ADAMS ML numbers for all referenced documents?

Comment [SW]: Add titles for Brad and Tony?

301-415-17XX

Based on ~~the~~ its review ~~documented in the attached report~~, the ~~panel~~ Panel concludes that the staff positions taken to support the compliance backfit finding represent new and different staff views on how to address potential pressurizer safety valve failures following water discharge. Although these staff positions are well-intentioned and conservative approaches that could provide additional safety margin, they do not provide a basis for a compliance backfit. In the absence of a failure of the pressurizer safety valve to reseat, the concerns articulated in the backfit related to event classification, event escalation, and compliance with 10 CFR 50.34(b) and General Design Criteria 15, 21, and 29 are no longer at issue.

The Panel notes, as did a member of the earlier NRR backfit appeal panel, that this issue ~~is generic in nature~~ appears to have generic applicability, and is not specific to Byron and Braidwood. The Panel believes that resolution of this issue would have benefited from consideration of the generic nature of the issue through the ~~application of~~ appropriate NRC~~'s~~ ~~generic issue~~ processes.

Your June 22, 2016 memorandum asked the ~~panel~~ Panel to answer five questions. These questions and the ~~panel's~~ Panel's responses follow:

1. Were the approvals based on a mistake? If so, what was the mistake and what are the implications for Braidwood and Byron?

   Answer: The 2001 and 2004 license amendments were based on reasonable and well-informed engineering judgment of the NRC staff, not a mistake.

2. What is the known and established standard for water qualification of pressurizer safety valves?

   Answer: The standard in place in 2001 and 2004 and at present is that ~~the probability of~~ failures of ~~a~~ passive pressurizer safety valves to reclose need not be assumed to occur following water discharge if the likelihood~~after passing water~~ is sufficiently small, based on well-informed staff engineering judgment~~, that it may be excluded from consideration in a deterministic analysis~~.

3. What is the known and established standard for progression of postulated events between categories of severity? Include a discussion of Regulatory Issue Summary 2005-29, "Anticipated Transients that Could Develop into More Serious Events," dated December 14, 2005, and the draft Revision 1 that was issued for public comment in 2015.

   Answer: For Byron and Braidwood, the standard for progression of postulated events between categories of severity is set forth in the Updated Final Safety Analysis Report (UFSAR), as described in the staff's October 9, 2015 backfit imposition letter. The Panel supports the staff's view that non-escalation (from ANS category II to ANS category IV) is a known and established standard applicable to Byron and Braidwood. However, this event progression standard does not establish specific standards for valve qualification. Therefore, it is not the basis for a compliance backfit given this set of facts. Regulatory Issue Summary 2005-29 and its draft Revision 1 do not alter this conclusion.

4. Does the current licensing basis for Braidwood and Byron comply with the applicable regulations? Is it adequate to provide protection to public health and safety?

Comment [SW]: Moved to the new sentence above. (I could go either way, but like closing the loop on the Exelon/NEI offer.)

Comment [SW]: Include a reference to Tony's memo somewhere?

Comment [SW]: Should we be a bit more specific about what "this issue" is? NRR's comments on our preliminary findings indicate some lack of clarity about what the issue is.

Comment [SW]: I don't think we're talking about the actual Generic (capital G) Issues (capital I) process, or are we? The revision leaves this as a possibility, but allows other options. See also, Michael's comment below.

Comment [MAS]: It seems appropriate to answer the 5 questions in the cover memo. Right now, they begin on page 11 of the report.

My proposed answers are based on our preliminary findings. I don't have a response to question 5.

Comment [SW]: In discussion with Brad Fewell at the UWC, he reiterated Exelon's position that there was no mistake, rather the staff has "reinterpreted" its positions.

Answer: The ~~panel~~ Panel concludes that the current licensing bases for Braidwood and Byron do comply with the applicable regulations based on the UFSAR analyses which the staff found acceptable through a reasonable and technically sound evaluation using appropriate Commission safety standards. The ~~panel~~ Panel also concludes that there is reasonable assurance of adequate protection of the public health and safety.

5. Given that Exelon suggests that the NRC pursue a cost-justified substantial safety enhancement backfit, what is the contribution to overall plant risk of the current configuration at Braidwood and Byron?

Answer: ~~An~~ The analysis performed for the panel by ~~the Office of Nuclear Regulatory Research (~~RES) provides insights on the risk significance of the sequence at issue. This analysis suggests that an inadvertent ECCS actuation sequence, assuming that pressurizer overfill leads to a small loss-of-coolant accident, contributes approximately 1 percent of the total internal events core damage frequency (CDF). If the backfit were implemented such that pressurizer overfill were always prevented, the CDF reduction is estimated at 1.5E-07 per year. ~~Different initial and final~~Less conservative assumptions ~~conditions~~ than these extremes would provide a smaller risk benefit through the backfit.

The Panel is aware of and sensitive to two important issues related to this question. First, NRR, not the appeal Panel, is responsible for any decisions on alternative application of the backfit rule to this issue. Second, the Panel does not wish to imply that "the contribution to plant risk" should be seen as the only measure of enhanced safety. For example, defense-in-depth has a recognized role and value in the regulatory process.~~TBD~~

The ~~panel's~~ Panel's findings therefore support the Exelon backfit appeal, and we recommend that you direct NRR to:

- w~~W~~ithdraw its compliance backfit finding,
- v~~V~~erify (e.g., through letter, meeting, owners group activity) that all PWRs have resolved this technical issue in a reasonable manner, and
- r~~R~~e-evaluate the matters discussed in Regulatory Issue Summary 2005-29 and its draft Revision 1 through ~~a more~~ the appropriate generic process to avoid the inappropriate or inadvertent imposition of backfits.

In the course of its activities, the ~~panel~~ Panel has developed several insights relevant to the backfit process and the use of generic processes to address potential safety issues. The ~~panel~~ Panel plans to share these insights with the CRGR for ~~their~~ its use in addressing your June 9, 2016, tasking related to implementation of agency backfitting and issue finality guidance. The Panel also identified other lessons from its review of the NRC evaluation of the performance of pressurizer safety valves for Braidwood, Byron, and other nuclear power plants that are identified in the attached report.

Finally, the Panel would like to recognize the cooperation of the NRR and OGC staff during this effort, and the timely and responsive efforts of RES in providing the comprehensive and useful risk analyses requested by the Panel.

Comment [SW]: I suggest that we add the RES analysis to the second paragraph of the memo, where we describe the scope of what we reviewed, and make the conforming editorial change noted here.

Comment [MAS]: This could be taken to mean "Generic Issues" process.

Comment [ST]: We should indicate that the report includes lessons from our review.

The ~~panel~~ Panel is available to respond to any questions or provide any other assistance needed.

[Patti to add concurrence page]

From:	West, Steven
To:	Clark, Theresa
Cc:	Holahan, Gary; Scarbrough, Thomas; Spencer, Michael; West, Steven
Subject:	Panel report comments
Date:	Wednesday, August 17, 2016 11:47:10 AM
Attachments:	Backfit Appeal Panel Report (MASTER) WEST 2016 08 17 1130.docx

I'm going to send my comments in chunks. Here are my comments on section 1 and a proposal for a new section 2.

Steve

Report of the Backfit Appeal Review Panel Chartered by the Executive Director of Operations to Evaluate the June 2016 Exelon Backfit Appeal

August 2016

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1 BACKGROUND

On June 22, 2016, in accordance with NRC Management Directive (MD) 8.4, "Management of Facility-specific Backfitting and Information Collection," the NRC Executive Director for Operations (EDO) established a Backfit Appeal Review Panel (Panel) to review the appeal by Exelon Generation Company, LLC (Exelon or the licensee) of the U.S. Nuclear Regulatory Commission (NRC) staff's determination that a backfit is necessary at Braidwood Station, Units 1 and 2 (Braidwood) and Byron Station, Units 1 and 2 (Byron), as well as the NRC staff's application of the compliance backfit exception provided in Title 10 of the *Code of Federal Regulations* (10 CFR), Section 50.109, "Backfitting."

Comment [SW]: Spell out first use. May also want to add some context. I think the letter has some that can be brought over.

This backfit determination is documented in an October 9, 2015, letter (Agencywide Documents Access and Management System (ADAMS) Accession No. ML14225A871). The letter describes the NRC staff's review of licensing basis documents for Braidwood and Byron ~~(Agencywide Documents Access and Management System (ADAMS) Accession No. ML14225A871)~~. The NRC staff determined that Braidwood and Byron were not in compliance with the plant-specific design bases and several NRC regulations:

- General Design Criterion (GDC) 15, "Reactor c~~C~~oolant S~~s~~ystem d~~D~~esign," in 10 CFR Appendix A, "General Design Criteria ~~(GDCs)~~ for Nuclear Power Plants"
- GDC 21, "Protection S~~s~~ystem R~~r~~eliability and ~~T~~testability"
- GDC 29, "Protection a~~A~~gainst a~~A~~nticipated ~~O~~operational ~~O~~occurrences"
- Paragraph (b) of 10 CFR 50.34, "Contents of applications; technical information"

Specifically, the NRC staff determined that Braidwood and Byron ~~were determined~~ do not ~~to~~ comply with provisions for ensuring that ANS Condition II events (analyses of inadvertent operation of the emergency core cooling system (IOECCS), malfunction of the chemical and volume control system (CVCS), and inadvertent opening of a pressurizer safety or relief valve) do not progress to more serious ANS Condition III events following water relief through certain valves. The NRC staff acknowledged that the NRC staff position differed from a previous staff position documented in a May 4, 2001, safety evaluation (SE) supporting a stretch power uprate (ADAMS Accession No. ML033040016). However, the NRC staff determined that the backfitting was justified under the compliance exception in 10 CFR 50.109(a)(4)(i). The licensee was directed to take action to resolve the non-compliance.

On December 8, 2015, the licensee appealed the NRC staff's decision stating its disagreement with the NRC's conclusion that the compliance exception to the backfit rule applies in this case, and that the NRC has twice approved the underlying analysis (ADAMS Accession No. ML15342A112). The referenced approvals were an August 26, 2004, license amendment associated with pressurizer safety valve (PSV) setpoints (ADAMS Accession No. ML042250531) and the above-referenced license amendment associated with a stretch power uprate. In a letter dated May 3, 2016, the NRC responded to the licensee's appeal and reaffirmed its decision that the backfit per the compliance exception provisions of 10 CFR 50.109(a)(4)(i) is appropriate (ADAMS Accession No. ML16095A204).

On June 2, 2016, the licensee again appealed the NRC staff's decision (ADAMS Accession No. ML16154A254), this time to the EDO. The purpose of this report by the Backfit Appeal

Review Panel is to provide information and recommendations to support the decision of the EDO.

1.1

1.1 Conduct of the Panel's Review

In order to establish a technically sound, well informed, and legally defensible basis for its recommendations, the Backfit Appeal Review Panel undertook a review of the relevant documents in this case. This included the licensee and NRC staff letters mentioned above; the 2001 power uprate and the 2004 license amendment; and a June 16, 2016, letter from the Nuclear Energy Institute (NEI) supporting the Exelon backfit appeal (ADAMS Accession No. ML16208A008~~ML16027A352~~). The Panel also reviewed many other related documents, which fall into ~~six~~ five broad categories:

- The Backfit Rule (10 CFR 50.109), related court actions, and Commission and staff guidance on application of the Backfit Rule
- Docketed communications for Byron and Braidwood (license amendment requests (LARs) by the licensee, NRC-issued license amendments, NRC requests for additional information (RAIs), licensee responses, meeting summaries, NRC ~~Safety Evaluations~~SEs, and the licensee's Updated Final Safety Analysis Report (UFSAR)) over the period of ~~1997~~ 1982 to the present
- NRC guidance relevant to the analysis of IOECCS events (Standard Review Plan (SRP) Section 15.0, "Introduction – Transient and Accident Analyses"; Section 15.5.1 – 15.5.2, "[IOECCS] and [CVCS] Malfunction that Increases Reactor Coolant Inventory"; and Section 15.6.1, "Inadvertent Opening of a PWR [Pressurized Water Reactor] Pressurizer Pressure Relief Valve or a BWR [Boiling Water Reactor] Pressure Relief Valve") over the period of 1981 to the present
- Westinghouse Nuclear Safety Advisory Letter (NSAL) 93-013, dated June 30, 1993~~,~~; ~~and~~ its Supplement 1, dated October 28, 1994; and docketed communications regarding actions taken by other licensees in response to Westinghouse NSAL-93-013
- ~~Docketed communications regarding actions taken by other licensees in response to Westinghouse NSAL-93-013~~
- The history of NRC and industry activities related to power operated relief valves (PORVs), their block valves, and PSVs (including Three Mile Island (TMI) action items II.D.1, II.D.3,II.G.1, II.K.3 documented in NUREG-0737, "Clarification of TMI Action Plan Requirements," as well as and Generic Letter 89-10 and its supplements), ~~related~~ Electric Power Research Institute (EPRI) valve testing, and operating experience (NUREG/CR-~~XXXX~~7037)

In addition to the document review, the Panel had the benefit of meetings with the Office of Nuclear Reactor Regulation (NRR) (both the Division of Safety Systems and the Division of Engineering), the Office of the General Counsel (OGC), and the NRC Committee to Review Generic Requirements (CRGR). Both Exelon (Bradley Fewell) and NEI (Tony Pietrangelo) declined offers for a public meeting, but indicated a willingness to provide information if the Panel identified the need.

Comment [CT]: This is the January letter, not the June one. Corrected (and will re-update when adding references).

Comment [CT]: Was defined on previous page.

Comment [SW]: Theresa, please consider making conforming changes throughout to reflect any of my comments noted in the cover letter that were accepted. Thanks.

At the request of the Panel, the Office of Nuclear Regulatory Research (RES) conducted risk analyses based on the NRC's Standardized Plant Analysis Risk model for [Byron]. These analyses informed the Panel's response to the question from the EDO regarding the risk significance of the relevant accident sequences.

Comment [MAS]: We should fully cite the statements here that we take issue with later, with explicit references to single failure and ASME.

1.2 Proposed Compliance Backfit and Exelon Appeals

In a letter dated October 9, 2015, the NRC staff informed Exelon that it had determined that Braidwood and Byron are not in compliance with GDC 15, 21, and 29; 10 CFR 50.34(b); and the plant-specific design bases that were expected to demonstrate there will be no progression of Category II events into Category III events. The NRC staff stated that based on its review of Braidwood and Byron UFSAR Sections 15.5.1, 15.5.2, and 15.6.1, the UFSAR predicts water relief through a valve that is not "qualified" for water relief. Therefore, the NRC staff concluded that the UFSAR does not contain analyses that demonstrate that the plants' structures, systems, and components (SSCs) will meet the design criteria for Condition II faults as stated in Braidwood and Byron UFSAR ~~Chapter~~ Section 15.0.1.2. Based on the Backfit SE attached to its letter, the NRC staff found that the licensee must take action to resolve the non-compliance.

The Backfit SE covers three accident analyses in Chapter 15 of the Braidwood and Byron UFSAR: (1) IOECCS; (2) CVCS malfunction that increases reactor coolant inventory; and (3) inadvertent opening of a pressurizer safety or relief valve (IOPORV). The NRC staff note~~s~~d that each Condition II event must be shown to meet the following:

1. No fuel damage

2. No overpressure of the reactor coolant system or main steam system

3. No progression into an event of a more serious category without another independent fault

Regarding IOECCS, the NRC staff state~~s~~d in Section 3.1.2.1 of the Backfit SE that use of the block valve to isolate a stuck-open PORV ~~is~~was unacceptable. The NRC staff state~~s~~d that Westinghouse recommended this approach in 1993 and that the NRC staff rejected this approach in 2005 (RIS 2005-~~029~~-29).

In Section 3.1.2.4 of the Backfit SE, the NRC staff state~~s~~d that the Braidwood and Byron IOECCS analysis depends on water relief through the PSVs. The NRC staff fault~~s~~d the licensee for "not [appl[ying] the single-failure assumption" and state~~s~~d that the following information ~~is~~was necessary to support water qualification of the PSVs:

1. In accordance with the American Society of Mechanical Engineers (ASME) *Boiler and Pressure Vessel Code* (BPV Code), Section III, provide the ~~Provide~~ original Overpressure Protection Report defining operating conditions and required relief capacities, and manufacturer's certification and test results

2. In accordance with the ASME *Code for Operation and Maintenance of Nuclear Power Plants* (OM Code), provide ~~Provide~~ inservice test history for PSVs, including water and steam tests, or provide correlation test for alternative test fluid.

Regarding CVCS malfunction, the NRC staff state~~s~~d in Section 3.2 of the Backfit SE that the licensee ha~~s~~d not provided an analysis for the CVCS malfunction that increases reactor coolant inventory that demonstrates the plants' ability to meet the requirements of a Condition II event.

Regarding IOPSRV, the NRC staff statesd in Section 3.3 of the Backfit SE that the licensee hasd not provided an analysis for the IOPORV that extends long enough into the transient to demonstrate the event wouldill not transition from a Condition II event to a Condition III event.

In the Backfit SE, the NRC staff referenced Millstone (1998) and Callaway (2000) ~~LARs~~ license amendments (ML011800207 and ML003719636) as examples of licensees upgrading PORVs for water relief; a Beaver Valley (2004) extended power uprate (EPU) requests (ML061720376ML?) as an example of qualifying PORVs for water relief; and Turkey Point and St. Lucie Unit 2 EPU ~~requests~~ amendments (ML11293A359 and ML12235A463) as additional precedent in support of the backfit decision.

On December 8, 2015, Exelon appealed the NRC backfit decision. Exelon assertsed that the NRC hasd not justified invoking the compliance exception to the backfit rule. Exelon statesed that the NRC approved its IOECCS analysis in the 2001 stretch power uprate and the 2004 PSV setpoint amendment.

On May 3, 2016, the NRC staff provided its review of the backfit appeal. The NRC staff stated that the previous approvals were inconsistent with the Agency's general position on the known and established standard at issue, in this case the progression of Condition II events to higher level events. The NRC staff stated that the fact that the NRC staff ~~had some awareness~~ aware of references to EPRI reports on the ability of these non-water qualified PSVs to reseat in certain circumstances is not sufficient to support the licensee's position.

On June 2, 2016, Exelon stated that the NRC hasd misidentified the "known and established standard" at issue as the prohibition of Condition II events progressing to Condition III events. Exelon assertsed that the standard in question concerns what is necessary to "qualify" valves for water discharge. Exelon contends that this standard is the EPRI testing and analysis, and that the NRC has agreed that Braidwood and Byron meet this standard. Exelon also contends that the change in NRC staff position on prior approvals is not a mistake of fact, but rather a new or modified interpretation of compliance with NRC requirements, for which use of the compliance exception provided for in the Backfit Rule is not appropriate.

1.3 Backfit Rule and the Compliance Exception

Backfitting is defined by 10 CFR 50.109(a) as:

> ... the modification of or addition to systems, structures, components, or design of a facility; or the design approval or manufacturing license for a facility; or the procedures or organization required to design, construct or operate a facility; any of which may result from a new or amended provision in the Commission's regulations or the imposition of a regulatory staff position interpreting the Commission's regulations that is either new or different from a previously applicable staff position ...

Unless one of three specified exceptions apply, the NRC may impose a backfit only if it performs a backfit analysis in accordance with 10 CFR 50.109(a)(2) and determines in accordance with 10 CFR 50.109(a)(3) "that there is a substantial increase in the overall protection of the public health and safety or the common defense and security to be derived from the backfit and that the direct and indirect costs of implementation for that facility are justified in view of this increased protection."

Comment [CT]: Confirm when updating references that these are the amendment packages (all in this paragraph)

Comment [CT]: Confirm when updating references that this is the amendment package.

Comment [SW]: Reference ML

Comment [SW]: Reference ML

Comment [SW]: Not sure what we are going for as far as tense goes, but we should be consistent throughout. Since Gary's mother is an English teacher, I defer to him; unless it's a legal thing. In either case, I'm going to quit editing this.

Section 50.109(a)(4) sets forth the three exceptions to the requirements of 10 CFR 50.109(a)(2) and (a)(3). The first exception, the compliance exception, applies if the "modification is necessary to bring a facility into compliance with a license or the rules or orders of the Commission, or into conformance with written commitments by the licensee." The second and third exceptions relate to actions necessary to ensure adequate protection or to actions that involve defining or redefining adequate protection.

The Commission explained its intended application of the compliance exception in the Statements of Consideration (SOC) accompanying the 1985 final rule amending 10 CFR 50.109 (Volume 50 of the *Federal Register* (FR), page 38103):

> The compliance exception is intended to address situations in which the licensee has failed to meet known and established standards of the Commission because of omission or mistake of fact. It should be noted that new or modified interpretations of what constitutes compliance would not fall within the exception and would require a backfit analysis and application of the standard.

In the same SOC (page 38102), the Commission acknowledged that staff interpretations of rules are not legally binding, but the Commission also stated that "staff interpretations of broadly stated rules are often necessary to give a rule effect and in some instances may be a causal factor in initiating a backfit."[1]

By its terms, the compliance exception applies to actions necessary for compliance with rules, licenses, and orders, or for conformance with written commitments.[2] Also, the Commission explicitly acknowledged the importance of staff interpretations of rules in the regulatory process. Thus, the Panel understands the term "known and established standard" to include standards established in rules, licenses, orders, and written commitments, and NRC interpretations of rules ~~that might be established through an appropriate generic issue process and announced in a variety of forms~~. Some standards may be broad-based, while others may apply only to a limited number of plants. As stated in NUREG-1409, "[i]nformal or formal communications to one licensee are not official positions to all licensees. ... Orders, licenses, and written commitments are applicable only to a particular licensee."

The failure to meet a known and established standard is grounds for a compliance backfit if this failure is due to "omission or mistake of fact." Thus, if a licensee obtains NRC approval of an alternative to a specific standard set forth in guidance, that standard and guidance could not be used to support a compliance backfit unless the NRC's approval of the alternative was based on an omission or mistake of fact. "Known and established standards" are to be distinguished from "new or modified interpretations of what constitutes compliance," which do not fall within the compliance exception. The Panel understands the term "new or modified interpretations" to include situations where the NRC has, in effect, "changed its mind" on how to interpret the language of a requirement or on how much assurance is necessary to conclude that the

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[1] The 1985 backfit rule was vacated by a Federal court on grounds unrelated to the compliance backfit exception. See *Union of Concerned Scientists v. U.S. Nuclear Regulatory Com'n*, 824 F.2d 108, 119-20 (D.C. Cir. 1987). In 1988, the Commission amended the backfit rule (53 FR 20603) to address the court's concerns, but did not change the 1985 rule's compliance exception provision. Thus, the quoted statements from the 1985 rule are the applicable expression of Commission intent regarding compliance backfits.

[2] NUREG-1409, Backfitting Guidelines, defines written commitments broadly to include the "final safety analysis report, licensee event reports, and docketed correspondence, including responses to NRC bulletins, generic letters, inspection reports, or notices of violation and confirmatory action letters."

requirement is met. Levels of assurance might be established in terms such as acceptable probabilities or consequences, conservative assumptions, or sufficient margin.

Additional background information on the Backfit Rule and the compliance exception is provided in Appendix A to this report.

1.4 A Brief History of ~~PORV~~ Power-Operated Relief Valve and ~~PSV~~ Pressurizer Safety Valve Issues

Appendix B to this report provides a summary of the NRC and industry's testing, evaluation, and other consideration of PORVs and PSVs since ~~issues were identified following~~ the TMI Unit 2 (TMI-2) accident in 1979. This historical review provides context for discussion of valve "qualification" in the Backfit SE. It also provides the basis for the Panel's conclusions regarding the "known and established standard" for "qualification" in the context of the TMI ~~A~~action ~~p~~Plan item and subsequent activities, as well as how it should be interpreted in the Byron and Braidwood licensing basis.

As noted above, the issue of the applicability of the Single Failure Criterion to PSVs is one of the key issues in determining whether the licensee is in compliance with NRC regulations. The Panel have expended considerable effort in searching for an answer to what appears to be a simple question, "Are PSVs active components subject to the Single Failure Criterion or passive components exempt from it?" NRR staff have taken the position that PSVs have consistently been treated as active components.

In the Panel's evaluation of the treatment of PSV failure potential (sections 2.1 – 2.11 below), an historical perspective is provided. In general, the Panel found that different organizations have defined "passive" components differently; almost always as not needing external power; usually as not needing an external actuator (e.g. signal); sometimes as not involving any mechanical motion (e.g. of a valve disc); and sometimes as not involving any motion, neither fluid nor mechanical motion (e.g. piping).

The introduction to the General Design Criteria (GDC) (10 CFR 50 Appendix A) and the related footnote define applicability of the single failure criteria in terms of electrical versus fluid systems, and active versus passive components. Neither the GDC nor NRC guidance define which characteristics of passive components are necessary to make a component exempt from the single failure criterion. Some examples are clear: pipes are passive components; pumps and motor-operated valves are active components; check valves are passive components (except in low-differential pressure passive systems, and some containment isolation applications). PSV are conservatively sized with sufficient margin to accommodate a single failure although the single failure criterion is almost never discussed or applied in accident analyses. The Byron and Braidwood UFAR states that "adequate overpressurization protection is provided by the three installed safety valves." Neither the UFSAR system descriptions nor the safety analyses discuss PSV any potential safety valve failures or their consequences.

Most relevant for the current issue, the Byron and Braidwood UFSAR analyses of over-pressure events (e.g. loss of load, loss of feedwater) do not apply the single failure criterion to cause a PSV to stick open when opening on steam flow. In addition, the UFSAR Feedwater System Pipe Break analysis (Chapter 15.2.8) does not apply the single failure criterion to cause a PSV to stick open either during steam discharge or during water discharge. A survey of other Westinghouse-designed plants showed that this treatment of PSV valve performance during

Comment [SW]: This is bound to raise questions; can we provide references that show disparate results.

Comment [SW]: I don't know if it is germane to what we are looking at, but license renewal essentially requires that all SSCs be categorized as either passive or active. The categorizations are used to establish whether or not aging management reviews and programs are needed. For completeness, should we see how PSVs have been categorized in LR?

Comment [SW]: Reference.

anticipated operational occurrences (AOO or ANS category II events) and postulated accidents (or ANS category IV events) has been consistent and without any identified exceptions.

1.5 ~~Throughout this report, the Byron and Braidwood PSVs are treated as passive components. This treatment is consistent with historical practice and American Nuclear Society (ANS), Standards Committee, Glossary of Definitions and Terminology (latest edition May 19, 2016).~~

1.5 History of Westinghouse NSAL and Related Activities

Appendix C to this report provides a review of the issues identified by Westinghouse in NSAL-93-13 and its Supplement 1, how various licensees responded to these issues, and how the NRC was involved in reviewing and approving these actions. This review provides the basis for the Panel's conclusions related to the approach taken by Byron and Braidwood to address these issues in their licensing basis, as well as on the "known and established standard" for event escalation from ANS Condition II to Condition III.

2 DISCUSSION

The compliance exception to the Backfit Rule is intended to address failures to meet known and established Commission standards because of omission or mistake of fact. New or modified interpretations of what constitutes compliance do not fall within the exception. The Panel reviewed and evaluated all the available information to determine if, in 2001 and 2004, there was a known and established standard of the Commission relating to the potential of PSVs to fail following water discharge during IOECCS events.

In addition, the Panel considered the issue of ~~a~~ "known and established standards of the Commission" as it relates to "event escalation." In its May 3, 2016, denial of the compliance backfit appeal, the NRC staff stated that "the October 9, 2015, backfit showed that the approvals at issue for Braidwood and Byron were inconsistent with the Agency's general position on the known and established standard at issue, in this case the progression of Condition II events." The Panel recognizes that the "non-escalation" position, although not included in NRC regulations, is widely referenced in reactor licensing bases as an approach for addressing anticipated operational occurrences (AOOs) and postulated accidents as articulated in the GDCs. The non-escalation position is incorporated in Section 15.0.1.2 of the Byron and Braidwood UFSAR as "By definition, these faults (or events) do not propagate to cause a more serious fault, i.e., Condition III or IV events."

Neither Exelon nor the Panel disputes that the non-escalation position is now, and was in 2001 and 2004, a part of the licensing basis of both Byron and Braidwood. The Panel supports the staff's view that non-escalation (from ANS category II to ANS category IV) is a known and established standard applicable to Byron and Braidwood. However, the Panel agrees with Exelon that the fundamental issue is not the non-escalation issue, but the appropriate standard for PSV water relief. In the absence of a PSV failure to reseat, the concerns articulated in the backfit related to event classification, event escalation, and compliance with 10 CFR 50.34(b) and GDCs 15, 21, and 29 would no longer be at issue.

As explained below, the Panel concludes that in 2001 and 2004 and at present, the known and established standard of the Commission is that the failures of passive safety valves need not be assumed to occur following water discharge if the likelihood is sufficiently small, based on well-informed staff engineering judgment. During the Exelon power uprate review in 2001 and the

Comment [SW]: I agree with Michael's comment below that it's taking a lot of reading before getting to the Panel's conclusions. I recommend a new Section 2 that provides an overview of the conclusions. I did this in a DPO report and it went over well. I don't have that report with me, but if you're interested and want to see what it looks like before Monday, see it you can get from OE the DPO report on Watts Bar flooding completed in 2012 or 2013.

Comment [MAS]: This is already page 7 and the panel's conclusion hasn't been stated. I think there needs to be a brief discussion of the Panel's conclusion and this seems an appropriate point. I took most of this from the draft preliminary conclusion, with updates to the statement of the inferred standard.

review of a later valve setpoint amendment in 2004, the staff exercised reasonable and well-informed engineering judgment when concluding that the PSVs were unlikely to stick open (i.e., fail to reseat). The non-escalation position does not establish specific standards for valve qualification, so the non-escalation position, standard alone, provides no basis for rejecting the licensee's reliance on EPRI valve testing. Further, the 2015 backfit identifies no omissions or mistakes in the EPRI testing or in the licensee's or the staff's earlier evaluation of this testing. Therefore, the Panel concludes that the position on valve qualification in the Backfit SE is a new or modified interpretation of what constitutes compliance.

Comment [CT]: Michael – your markup appeared to delete "Part" and the comma, but they weren't in this version, so I wonder if you were really adding them. I like them but could go either way.

The Panel's evaluation relative to treatment of PSV failure potential includes an assessment of the following references.

2.1 1971 10 CFR Part 50 Appendix A, Footnote 2

In 1971, the Atomic Energy Commission published the General Design Criteria (10 CFR Part 50, Appendix A) which had been under development since 1968. The introduction to Appendix A addresses "Single Failure" in the section on Definitions and Explanations. The paragraph on single failures includes a footnote stating: "The conditions under which a single failure of a passive component in a fluid system should be considered in designing the system against a single failure are under development." ([emphasis added).]

2.2 1977 SECY-77-439

In response to several staff concerns and differing views on the subject of application of the single failure criterion, the Acting Director of NRR issued a Commission paper "[t]o inform the Commission of the present status and future use of the Single Failure Criterion as a tool in the reactor safety process." In part, that paper addressed the application of the single failure criterion to passive components in fluid systems, stating that "[a]pplication of the [single failure] concept is complicated by the interrelationships between the various fluid and electrical systems and their supporting auxiliaries in a nuclear power plant. Furthermore, there is a need to stipulate the events and associated assumptions which must be considered during application of the Single Failure Criterion."

SECY-77-439 specifically spoke to how "additional passive failures"—that is, failures in addition to the initiating event—had been and should be addressed, stating (emphases added):

> During subsequent years [since the single failure footnote quoted above was published] staff assumptions regarding the nature of passive failures which should be considered have not been completely consistent and there has been some disagreement. However, on the basis of the licensing review experience accumulated in the period since 1969, it has been judged in most instances that the probability of most types of passive failures in fluid systems is sufficiently small that they need not be assumed in addition to the initiating failure in application of the Single Failure Criterion to assure safety of a nuclear power plant. ~~[emphasis added]~~

The paper also stresses the use of engineering judgment relating to the probability of component failure and does not suggest that valve "certification" or "qualification" in accordance with ~~American Society of Mechanical Engineers~~ (ASME) standards should be invoked as the basis for such decisions.

2.3 1979 TMI Action Plan item II.D.1

As an element of the TMI Action Plan (NUREG-0585 and NUREG-0737), the NRC staff required licensees to address the capability of relief and safety valves to perform their intended functions without failure. Specifically, Item II.D.1 states that "[PWR and BWR] licensees and applicants shall conduct testing to qualify the [RCS] relief and safety valves under expected operating conditions for design-basis transients and accidents." NUREG-0737 specified provisions for then-operating nuclear power plants and applicants for operating licenses and holders of construction permits to address the TMI ~~a~~Action Plan items, including Item II.D.1. NUREG-0737 stated, for the performance testing of relief and safety valves for Item II.D.1 that "[t]he testing should demonstrate that the valves will open and reclose under the expected flow conditions," with reference to planned EPRI testing and other generic industry test programs.

Although limited in scope, the EPRI test results did not identify any generic issues with PSVs or PORVs sticking open following water relief. The historical record shows that the word "qualify" in this TMI Action Plan item was not intended to refer to ASME valve certification or qualification. Instead, "qualify" was used in a less formal sense to refer to a reasonable judgment that the valve would open to relieve pressure and then reliably reseat. As referenced in NUREG-0737, the EPRI test program was the widely used approach to address TMI Action Plan Item II.D.1 at PWR nuclear power plants. In a letter dated July 27, 1982, the Westinghouse Owners Group submitted WCAP-10105 to the NRC to demonstrate the acceptability of the EPRI testing program for PSVs and PORVs in Westinghouse-designed PWRs.

2.4 1988 and 1990 NRC Letters on TMI Action Plan Item II.D.1 for Byron and Braidwood

A 1988 letter from the NRC staff to the licensee for Byron found the licensee's reliance on EPRI testing of PSVs to be acceptable. The SE states that the test program was designed "[t]o reconfirm the integrity of the overpressure protection system and thereby assure that the ~~General Design Criteria (~~[GDCs]~~)~~ are met." The SE contains no reference to or suggestion of a need to undertake additional activities such as certification in accordance with the ASME ~~*Boiler and Pressure Vessel (BPV) Code*~~BPV Code. In 1990, the use of the EPRI test program was also found similarly acceptable for Braidwood.

2.5 Westinghouse NSAL-93-013 and Supplement 1

In 1993, Westinghouse sent ~~Nuclear Safety Advisory Letter~~ NSAL-93-013 to operating nuclear power plants in response to its discovery that potentially non-conservative assumptions had been used in the licensing analysis of the IOECCS event. Westinghouse recommended that licensees determine if their Pressurizer Safety Relief Valves (PSRVs)[3] "are capable of closing following discharge of subcooled water." Westinghouse noted that the PSRVs might have been designed or "qualified" to relieve subcooled water. Westinghouse also noted that "licensees may have qualified these valves in compliance ~~with~~ to NUREG-0737, Item II.D.1." If the PSRVs were not designed or qualified for subcooled water relief, Westinghouse recommended that licensees reevaluate the IOECCS event with three possible options of (1) reducing emergency core

[3] The Westinghouse use of the term P~~R~~SRVs is technically incorrect, in that the valves in question should be designated as "Safety Valves" or "Pressurizer Safety Valves" as they are by the manufacturer, in the ASME BPV Code, and by the licensee. This difference in terminology is not significant to any of the findings or conclusions in this report.

Comment [CT]: Add references (highlighted text in header).

Comment [CT]: Defined above so put in brackets.

Comment [CT]: Defined above.

Comment [CT]: Actually says "to" in the NSAL.

cooling system (ECCS) flow used in the safety analysis, (2) using a less restrictive operator response time, or (3) crediting the use of one or more PORVs to help mitigate the accident.

Comment [CT]: Defined above

In Supplement 1 to NSAL-93-013, Westinghouse alerted licensees to potential reduced time for operator action if a positive displacement pump (a typical part of the CVCS) were in service, and to the need to qualify the PSRVs and the piping downstream of the PSRVs and PORVs if water relief from the pressurizer is predicted. Some licensees submitted license amendments that involved improvements to the PORVs and their circuitry to avoid water relief through the PSVs (e.g., Diablo Canyon in 1996, Salem in 1997, Millstone 3 in 1998, and Callaway in 2000). The NRC staff review and approval of those proposed improvements relied on engineering judgment relative to the various test information and PORV circuitry upgrades described by individual licensees. In 1997, Exelon submitted an ~~License Amendment Request (~~LAR~~)~~ for similar PORV improvements at Byron and Braidwood, but that request was later withdrawn.

As indicated below, the Panel's sampling review found two licensees, in addition to Byron and Braidwood, that chose to address this issue on the basis of safety valve capability to relieve water based on the testing done in response to TMI Action Plan Item II.D.1.

1994 SECY-94-084 Policy and Technical Issues... in Passive Plant Designs

In 1994, in preparation for the design certification reviews of passive reactor designs (e.g. AP1000, ESBWR), the staff presented nine issues to the Commission for policy decisions (SECY-94-084 March 28, 1994 Policy and Technical Issues Associated with the Regulatory Treatment of Non-Safety Systems in Passive Plant Designs (ML003708068)). Although safety valve categorization and performance requirements were not explicitly addressed, the paper does include an issue on "Definition of Passive Failure" and an extensive discussion on whether check valves are passive or active components and how they should be addressed in current plants and future passive designs.

SECY-94-084 recognizes the GDC and SECY-77-439 as establishing long-standing requirements and guidance in this area. The paper acknowledges that the industry (including EPRI and ANSI/ANS 58.9) have been inconsistent with respect to check valve failures, sometimes considering them failure as "active failure", sometimes as "passive failures". However, in SECY-77-439 the staff stated that the failure of a check valve to move to its correct position when required was a "passive failure". In addition SECY-94-084 states, "In licensing reviews, however, only on a long-term basis [i.e. long-term recirculation cooling following a LOCA] does the staff consider passive failures in fluid systems as potential accident initiators in addition to initiating events" and "For current plants, the NRC staff normally treats check valves, except for those in containment isolation systems, as passive devices during transients or design-basis accidents."
Furthermore, SECY-94-084 states "Redefining check valves as active components, subject to consideration for single active failures would cause these valves to be evaluated in a more stringent manner than that used in previous licensing reviews." The staff then recommended (and the Commission agreed in the on SRM-94-084 June 30, 1994) "The staff recommends that the Commission approve the staff's proposal to maintain the current licensing practice for passive component failures on the passive ALWR designs, and to redefine check valves, except for those whose proper function can be demonstrated and documented, in the passive safety systems as active components subject to single failure consideration."

The panel considers check valves and safety valves to be similar in that they both function through the motion of the valve disk under differential pressure with no external signal or motive power. The panel also recognizes that the ambiguity with respect to "passive" versus "active" component definitions and nomenclature exists for safety valves. However a survey of Westinghouse-designed plants indicates that PSVs are conservatively sized with sufficient margin to accommodate a single failure although the single failure criterion is almost never discussed or applied in accident analyses. In addition the UFSAR analyses of over-pressure events (e.g. loss of load, loss of feedwater) do not apply the single failure criterion to cause a PSV to stick open when opening on steam flow. Furthermore, the UFSAR Feedwater System Pipe Break analyses do not apply the single failure criterion to cause a PSV to stick open either during steam discharge or during water discharge.

2.7 Draft Standard Review Plan (SRP) Section 15.5.1 (1996)

SRP Section 15.5.1 (1996) includes extensive updated to the 1981 SRP, but neither version includes any discussion, criteria, or guidance on applying the single failure criteria or any other failure assumption to PSVs.

2.8 2001 – 2006 Power Uprate Reviews and License Amendments

As part of the 2001 power uprate review for Byron and Braidwood, the NRC staff approved the analysis of an IOECCS (UFSAR ~~Chapter~~ Section 15.5.1) that included pressurizer filling, PSV water discharge, ECCS termination, and PSV closure. In support of the 2015 backfit, the NRC staff suggests that the 2001 license amendment was predicated on the NRC's mistaken belief that the valves were ASME-qualified (certified). However, a review of the SE and associated RAIs shows that, in 2001, the NRC staff was well aware of the nature of the EPRI testing being relied on. The Panel did not find any evidence that the licensee claimed or the NRC staff believed that the valves were "qualified" in an ASME certification sense; rather, the record shows thorough consideration of the testing conducted on valves of the type installed at the plant and a technical judgment that this testing provided appropriate qualification.

The Panel's conclusion was confirmed via discussions with the individual who was the responsible Section Chief in the Reactor Systems Branch at the time. He informed the Panel that the 2001 license amendment was based on the exercise of staff engineering judgment and there was no discussion of ASME certification or qualification of valves. In addition, the NRC approved power uprates for other nuclear power plants that included NRC staff evaluation of water relief through PORVs or PSVs based on test information provided by individual licensees.

In 2001, the NRC granted a power uprate for Shearon Harris that included the operability of PORVs and PSVs during the discharge of subcooled water, referencing TMI Action Plan Item II.D.1. In 2006, the NRC granted a power uprate for Beaver Valley. The SE for this amendment ~~that~~ referred to RIS 2005-~~029~~-29 and found reasonable assurance that the PSVs would adequately discharge water and reseat following a spurious safety injection actuation, based on the EPRI test data from 1981 and an evaluation of the temperature of the liquid being discharged.

During the NRC evaluations of license amendments since the TMI-2 accident, the NRC staff has specified in some SEs that a PORV or PSV would be assumed to stick open if it was not qualified for liquid service. To address this concern, the NRC staff reviewed and accepted a variety of test information (including EPRI, Wyle, and vendor testing) from individual licensees for the capability of PORVs or PSVs to reseat following water relief. A specific requirement for

Comment [MAS]: I took this from the Summary. The Discussion section should support the Summary.

Comment [CT]: Placeholder to speak even further to the RAI response and SE, if warranted based on NRR comments.

Comment [CT]: Per Michael's comment, need to add a sentence why this item is relevant.

the PORVs or PSVs to be certified under the ASME BPV Code as capable of passing water and reclosing was not found in the reviewed sample of SEs.

In 2004, NRC issued a license amendment for ~~the~~ Braidwood and Byron ~~Stations~~ granting an adjustment to the PSV setpoints. In an RAI, the NRC staff requested that ~~Exelon~~ the licensee perform a quantitative analysis regarding the number of cycles during which the PSV would pass water and the temperature of the water being discharged. In its SE, the NRC staff concluded that the reanalysis was acceptable ~~to assure~~for assuring that the PSVs will remain operable following a spurious safety injection event.

2.9 2005 RIS 2005-~~029~~-29

In 2005, the NRC staff issued RIS 2005-~~029~~-29 "to notify licensees of a concern identified during recent reviews of power uprate [LARs]." The RIS addressed the manner in which some licensees acted in response to NSAL-93-013. The RIS was issued at the division level in NRR and did not include office-level review or concurrence. The RIS was not formally reviewed by CRGR. Although no documentation was found, it appears that the lack of a CRGR review stems from the assertions in the RIS such as these:

- "This RIS requires no action or written response and, therefore, is not a backfit under 10 CFR 50.109. Consequently, the NRC staff did not perform a backfit analysis."
- "This RIS is informational and pertains to a NRC staff position that does not depart from current regulatory requirements and practice."

A key statement in RIS 2005-~~029~~-29 is the following ~~(~~[emphasis added~~)~~]:

> The NRC staff's position is noted in the power uprate review standard, as follows: "For the [IOECCS] and [CVCS] malfunctions that increase reactor coolant inventory events: (a) non-safety-grade pressure-operated relief valves should not be credited for event mitigation and (b) pressurizer level should not be allowed to reach a pressurizer water-solid condition."

However, the ~~Power Uprate R~~cited review ~~S~~standard (RS-001 2003), which is explicitly limited to extended power uprates, states that "[t]he staff does not intend to impose the criteria and/or guidance in this review standard on plants whose design bases do not include these criteria and/or guidance. No backfitting is intended or approved in connection with the issuance of this review standard."

This intent of RS-001 to define and clarify the scope of power uprate reviews, but not impose new requirements or new interpretations of requirements ~~(i.e., "no backfit")~~ was confirmed in personal discussions with the NRR manager responsible for developing and issuing RS-001. Therefore, contrary to the RIS statement, neither the RS-001 review standard nor RIS 2005-~~029~~-29 documented "known and established standards of the Commission."

The Panel also notes that neither RIS 2005-~~029~~-29 nor its draft Revision 1 discuss water relief certification requirements in accordance with the ASME ~~BPV~~ Code. In fact, as stated above, the NRC issued a 2006 power uprate amendment for Beaver Valley in which the SE cited RIS 2005-29 and yet relied on the EPRI testing data to address the concern.

2.10 2005 SECY-05-0138

In August 2005, the NRC staff issued SECY-05-0138, "Risk-Informed and Performance-Based Alternatives to the Single-Failure Criterion." SECY-05-0138 presents a comprehensive history of the application of the single failure criterion including extensive discussion of the treatment of passive components in fluid systems. The paper acknowledges that "[o]ne particular issue identified in this project is the continued existence of the footnote to the definition of single failure in 10 CFR 50 Appendix A stating that the regulatory position on considering passive failures in fluid systems is under development." The paper quotes from SECY-77-439 (discussed above) and recognizes that in current practice, as in 1977, ~~that~~ "[p]assive failures in fluid systems are generally excluded from single-failure assessments."

SECY-05-0138 presents three alternatives for using a risk-informed and performance-based approach to address the single failure issue. The paper states that "[a]ll alternatives could include developing a position on single passive failures in fluid systems to replace the footnote now in 10 CFR Part 50 Appendix A definitions."

The paper also makes it clear that, with few exceptions, neither the NRC staff nor the Commission has established specific requirements relating to the treatment of passive component failures in fluid systems. The Panel believes the existence of this Commission paper, contemporaneous with discussions on potential safety valve failures (e.g~~.~~., ~~RIS~~ RIS 2005-~~029~~-29), makes it clear that no specific "established standards" on safety valve failures had been developed between 1977 and the time of the Byron and Braidwood ~~power uprate decisions~~license amendments in 2001 and 2004.

Standard Review Plan (SRP) Section 15.5.1 (2007)

SRP Section 15.5.1 (2007) states, "The pressurizer safety valves, too, may be assumed to reseat properly after having relieved water; but only if such valves have been qualified for water relief." However, this section does not reference ASME requirements for qualification.

2.12 NRR 2015 Compliance Backfit Finding and the Appeals

The NRR 2015 compliance backfit (October 9, 2015, letter to Exelon) is predicated on the following positions (emphases added):

- "water relief through a valve that is not qualified for water relief will cause that valve to stick in its fully open position"
- "the licensee ... has not applied the single-failure assumption"
- "nor have they provided ASME water qualification documentation for the PSVs ... the ASME ... original Overpressure Protection Report ... inservice test history ... including both water and steam tests"

The compliance backfit then argues that the IOECCS ~~(an AOO per the GDC definition and a Condition II event in the ANS classification system adopted in the Byron and Braidwood UFSARs)~~ would escalate to a more severe event. Such an escalation would be contrary to the Byron and Braidwood licensing basis (i.e., contrary to the ANS non-escalation position) and could be in non-compliance with the GDCs (as included in the Byron and Braidwood licensing basis) since an IOECCS with a stuck-open valve had not been analyzed and shown to meet to appropriate criteria for an AOO.

~~The Panel has reviewed the October 9, 2015, backfit letter to Exelon, the Exelon response and appeal letter of December 8, 2015 (referred to as the NRR appeal), and multiple documents associated with the NRR appeal, including:~~

~~the Exelon meeting presentation of March 7, 2016 in defense of their appeal~~

~~the NRR appeal panel meeting minutes (January 15, 2016; February 5, 2016; February 25, 2016; and March 15, 2016)~~

~~a memorandum documenting the input of one NRR appeal panel member (A Gody to M Bailey March 21, 2016)~~

~~the May 3, 2016, appeal response by the director of NRR, including the NRR appeal panel's report~~

~~As noted previously, Exelon appealed to the NRC EDO, and NEI provided a letter supporting this appeal; the Panel reviewed these documents as well.~~

Based on a review of all the relevant documents, and discussions with numerous parties involved in the original review and the compliance backfit ~~proposal~~, the Panel has developed a good understanding of the regulatory requirements and practices, the potential safety issues, and backfit rule obligations. The Panel has determined that the numerous, complex, and detailed regulatory and technical issues all depend on the answers to ~~three~~ two critical questions on valve performance, namely:

- Must the PSVs in question be assumed to fail given liquid water discharge because of the lack of ASME certification for water discharge?
- ~~May~~ Must the PSVs be assumed to fail in accordance with the GDC "single failure" requirements?

~~Are the Byron or Braidwood PSVs so likely to fail and to stick open, given liquid water discharge, that the NRC staff must assume their failure as a normal consequence of that condition and not as an "independent fault," but as a consequential failure?~~

Comment [CT]: Per Michael's comment – should confirm.

Comment [CT]: Proposing to delete based on Michael's comment – I think I agree that the first question covers it.

In the~~ir~~ October 9, 2015, letter to Exelon, the NRC staff indicates that "[o]ne assumption that is particularly important to the non-escalation criteria is that water relief through a valve that is not qualified for water relief will cause that valve to stick in its fully open position" [emphasis added]. The Panel concludes that this issue, the treatment of potential valve failure, is not only "particularly important," it is the critical issue upon which the compliance backfit hinges.

3 RESPONSE TO THE EDO QUESTIONS

In establishing the Panel, the EDO asked the panel to answer five specific questions, as well as evaluating the overall appropriateness of the October 2015 backfit. The answers to these questions are provided below.

3.1 Were the approvals based on a mistake? If so, what was the mistake and what are the implications for Braidwood and Byron?

In responding the question, the Panel has considered the differing views of the NRR staff and the licensee on this issue. Those positions are summarized below:

- In the May 3, 2016, letter to Exelon, the NRC staff claims that "[t]he NRC erred in approving a sequence of events that allowed the [IOECCS], [CVCS] malfunction, and inadvertent opening of a pressurizer safety or relief valve analyses in the 2001 and 2004 [SEs]" and "the NRC staff understood the PSVs to be qualified for water relief when, in fact, they were not."

- Exelon claims, in ~~their~~ the December 8, 2015, backfit appeal letter, that "the compliance exception requires more than simply asserting that the prior staff approvals were wrong—the NRC must demonstrate that the prior approvals were erroneous because of an omission or mistake of fact at the time of the approval. The NRC has not made that case here."

Comment [CT]: Note when doing references – Michael recommends placeholder titles (e.g., NRR backfit appeal ruling).

Comment [CT]: Note for Michael – I think we typically don't hyphenate adverbs (which was your suggestion).

The Panel concludes that, in 2001 and 2004, the NRC staff did not misunderstand the qualification status of the PSVs and that it was not a ~~categorical~~ mistake to undertake a review of or make a technically based safety finding on the likely successful performance of the valves. In the Panel's opinion, the actions of the Reactor Systems Branch in 2001 to reach out to the Division of Engineering/Mechanical Engineering Branch for expert technical review assistance was both appropriate and ~~noteworthy~~commendable. The NRC staff reviewers involved in the 2001 power uprate review were among the most experienced and senior reviewers in their areas of expertise. The NRC staff valve expert involved in the review was the agency's most knowledgeable individual on PSVs and the relevant ASME Code requirements, and was a nationally recognized expert. The Panel cannot agree that the NRC staff was misinformed, ill-informed, or in error, or that ~~they~~ it made incorrect or inappropriate decisions.

3.2 What is the known and established standard for water qualification of PSVs?

The Panel concludes that in 2001 and 2004 and at present, the known and established standard of the Commission is that the failures of ~~mechanical~~ passive s~~S~~afety v~~V~~alves need not be assumed to occur following water discharge~~–~~ if the likelihood is sufficiently small,~~—~~ based on well-informed staff engineering judgment~~—~~. No more detailed or prescriptive standard has been promulgated by the Commission.

3.3 What is the known and established standard for progression of postulated events between categories of severity?

For Byron and Braidwood, the standard for progression of postulated events between categories of severity is set forth in the UFSAR as described above. . The Panel supports the staff's view that non-escalation (from ANS category II to ANS category IV) is a known and established standard applicable to Byron and Braidwood. However, this event progression standard does not establish specific standards for valve qualification. Therefore, it is not the basis for a compliance backfit given this set of facts.

In answering this question, the Panel was also asked to include a discussion of RIS 2005-~~029~~-29 and the draft Revision 1 that was issued for public comment in 2015.

The Panel has reviewed the issue of "event escalation" as discussed in the compliance backfit and in RIS 2005-~~029~~-29 and the draft Revision 1. The Panel concludes that the IOECCS (an AOO per the GDC definition and a Condition II event in the ANS classification system adopted in the Byron and Braidwood UFSARs) would escalate to a more severe event if a PSV were to stick open, or if both a PORV stuck open and its block valve failed to close. Such an escalation would be contrary to the Byron and Braidwood licensing basis (i.e., contrary to the ANS non-escalation position) and could be in non-compliance with the GDC (as included in the Byron and Braidwood licensing basis), since an IOECCS with a stuck-open valve had not been analyzed and shown to meet to appropriate criteria for an AOO.

Comment [CT]: Michael's comment: I'm not seeing the connection to the RIS? This seems more related to answering the first part of the EDO's question. Does the RIS really change anything here? Is this really just based on the UFSAR?

Should we delete/revise/etc.?

3.4 Does the current licensing basis for Braidwood and Byron comply with the applicable regulations? Is it adequate to provide protection to public health and safety?

The Panel concludes that Braidwood and Byron do comply with the applicable regulations based on the UFSAR analyses, which the NRC staff found acceptable through a reasonable and technically sound evaluation using appropriate Commission safety standards.

3.5 Given that Exelon suggests that the NRC pursue a cost-justified substantial safety enhancement backfit, what is the contribution to overall plant risk of the current configuration at Braidwood and Byron?

The Panel requested the Office of Nuclear Regulatory Research to provide information and insights on the risk significance of the sequence at issue, to assure that the Panel's judgments were being made with a full understanding of their significance, and to assist in responding to the EDO question.

The RES study suggests that the most significant IOECCS sequence, assuming that all pressurizer overfill events lead to a small LOCA, contributes approximately 1 percent of the total internal event core damage frequency (CDF). In the report, the maximum benefit (CDF reduction) from a "perfect backfit" (i.e., always preventing pressurizer overfill) is estimated at 1.5E-07 per year. If the PSVs are not assumed to always fail following water discharge (consistent with the staff expert judgment in 2001) or a smaller improvement than a "perfect backfit" were considered, the risk-reduction benefit of implementing the backfit would be even smaller.

The Panel is aware of and sensitive to two important issues related to this question. First, NRR, not the appeal Panel, is responsible for any decisions on alternative application of the backfit rule to this issue (through the other categories of adequate protection or cost-justified substantial safety enhancement). Second, the Panel does not wish to imply that "the contribution to plant risk" should be seen as the only measure of enhanced safety. The issues of event classification and the non-escalation of events are essentially defense-in-depth concepts. Defense in depth has a recognized role and value in the regulatory process. The Panel is also aware that not every defense-in-depth feature has the same safety significance, and that the estimated risk significance (measured in core damage frequency) is very relevant.

Within the context suggested above, the Panel concludes that the contribution to overall plant risk is very small ~~(input from RES)~~.

4 SUMMARY AND CONCLUSIONS

The compliance exception to the Backfit Rule is intended to address failures to meet known and established Commission standards because of omission or mistake of fact. New or modified interpretations of what constitutes compliance do not fall within the exception. Therefore, to address the appeal of the proposed compliance backfit, the Panel has focused on determining if this case is most appropriately characterized as one in which the ~~NRC staff and the~~ licensee "failed to meet known and established standards of the Commission because of omission or mistake of fact," or rather as a case of a "new or modified interpretations of what constitutes compliance."

The NRC staff's compliance backfit argument depends on two separate determinations: the assumed failure of PSVs to reclose after passing water and the necessity of preventing "event escalation" (i.e., the position that "an incident of moderate frequency should not generate a more serious plant condition without other faults occurring independently"). For the backfit to be valid, both of these determinations must meet the above compliance backfit standard by involving failure to meet known and established standards of the Commission.

In the first of these determinations, the NRC staff's compliance backfit finding is based on the assumption in the SE that the PSV fails to reclose given the absence of "ASME water qualification documentation." ~~The compliance backfit asserts that staff was mistaken in using expert technical judgment and in reviewing the licensee's submittal documenting the EPRI valve test results (performed in response to an NRC requirement (TMI Action Item II.D.1) and previously evaluated and found acceptable by the NRC staff).~~

As indicated in the compliance backfit ~~proposal~~, the 2001 NRC staff SE for the Byron and Braidwood power uprate did involve a technical evaluation of safety valve capability and likely performance under water-discharge conditions rather than a simple assumption of a failure. The NRC response to the Exelon first appeal indicates that "the 2001 and 2004 approvals occurred because the NRC staff understood the PSVs to be qualified for water relief when, in fact, they were not."

The Panel has carefully considered these views and has reviewed the relevant documents including the NRC staff's RAIs and the licensee's responses, the NRC staff SE input, and the final staff SE written at the time of the 2001 power uprate review. 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 certification sense; rather, the record shows thorough consideration of the testing conducted on valves of the type installed at the plant and a technical judgment that this testing provided appropriate qualification.

The Panel concludes that the NRC staff did not misunderstand the qualification status of the PSVs and that it was not a ~~categorical~~ mistake to undertake a review of or make a technically based safety finding on the likely successful performance of the valves. In the Panel's opinion, the actions of the Reactor Systems Branch in 2001 to reach out to the Division of Engineering/Mechanical Engineering Branch for expert technical review assistance was both appropriate and ~~noteworthy~~commendable. The NRC staff reviewers involved in the 2001 power uprate review were among the most experienced and senior reviewers in their areas of expertise. The NRC staff valve expert involved in the review was the agency's most knowledgeable individual on PSVs and the relevant ASME Code requirements, and was a nationally recognized expert. The Panel cannot agree that the NRC staff was misinformed, ill-informed, or in error, or that ~~they~~ it made incorrect or inappropriate decisions.

In the Panel's opinion, three related technical and regulatory positions underpin the backfit:

- ASME water qualification (certification) documentation is required if a valve is to be assumed to reclose after passing water.
- ~~W~~water relief through a steam-qualified valve will cause that valve to stick in its fully open position.

Comment [CT]: Placeholder to add further discussion about new idea that the 2001 reviewer thought it had previously been approved to discharge water in this particular event (which it hadn't).

- PSVs are subject to a single-failure assumption.

None of these positions were "known and established standards of the Commission" in 2001 or 2004 for determining when it was appropriate to assume a failure of PSVs to reseat. In fact, they were not "known and established standards of the Commission" in 2005 (RIS 2005-29) or 2006 (Beaver Valley EPU) or 2007 ~~when similar decisions were made~~(SRP Section 15.1.1 update).

Moreover, two of these positions do not appear to be "established standards of the Commission" at present. The call for ASME certification first appears in the Exelon compliance backfit. The call for use of the single failure criterion first appears in proposed 2015 Draft Revision 1 to RIS 2005-~~029~~-29.

The Panel concludes that the standard in place in 2001 and 2004 and at present is simply that the failures of ~~mechanical~~ passive ~~S~~safety ~~V~~valves need not be assumed to occur following water discharge if the likelihood is sufficiently small, —based on well-informed staff engineering judgment. In earlier documents addressing this topic, beginning with NUREG-0737, ~~The standard is also that~~ the use of the word "qualified" or "qualification" implies ~~only~~ a general demonstration of capability, such as in the EPRI testing done in response to TMI Action Plan Item II.D.1. In light of ~~these~~ this standards, during the Exelon power uprate review in 2001 and the review of a later valve setpoint amendment in 2004, the NRC staff exercised reasonable and well-informed engineering judgment when concluding that the PSVs were unlikely to stick open (i.e., fail to reseat).

The Panel has concluded that the position on valve qualification in the 2015 backfit is a new or modified interpretation of what constitutes compliance in addressing potential PSV failures following water discharge. Although this position represents a well-intentioned and conservative approach that could provide additional safety margin, it does not provide a basis for a compliance backfit.

In the absence of a PSV failure to reseat, the concerns articulated in the backfit 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 Panel's findings therefore support the Exelon backfit appeal.

5 ADDITIONAL PANEL THOUGHTS

In addition to the specific finding relating to the backfit appeal, the Panel believes it is important to acknowledge, and for the NRC staff and licensee to appreciate, that water discharge through an PSV not specifically designed for such service is undesirable and should be minimized or avoided as a matter of conservative engineering and prudent operations. The Panel concludes this while fully aware that the event sequence being considered appears to be of little safety significance. Nonetheless, operator training and emergency procedures to terminate the event before pressurizer filling, as well as the use of PORVs rather than relying solely on PSVs, are clearly preferred prudent measures, whether they form the facilities' UFSAR licensing basis and are assumed in the accident analyses or not.

Comment [CT]: Added per Michael's comment – confirm this is what was meant.

Comment [CT]: Did not yet include Tom's paragraph as it merits more discussion and perhaps engagement with Exelon.

APPENDIX A: HISTORY OF THE BACKFIT RULE AND THE COMPLIANCE EXCEPTION

The Backfit Rule

Title 10 of the *Code of Federal Regulations* (10 CFR), Section 50.109, "Backfitting," was originally promulgated in 1970 (Volume 35 of the *Federal Register* (FR), page 5317). Because of perceived deficiencies in the rule, the U.S. Nuclear Regulatory Commission (NRC) substantially revised it in 1985 (50 FR 38097). The 1985 rule was challenged in court, and the U.S. Circuit Court for the District of Columbia (D.C. Circuit) vacated this rule in its entirety. The D.C. Circuit took this action because it concluded that the revised rule could be interpreted to allow the NRC to consider costs in defining or redefining what is required for adequate protection of the public health and safety. *Union of Concerned Scientists v. U.S. Nuclear Regulatory Com'n*, 824 F.2d 108, 119-20 (1987). In response, the NRC revised the Backfit Rule in 1988 (53 FR 20603) to remove any implication that costs could be considered in defining or redefining adequate protection. The 1988 revisions only differed from the 1985 rule to the extent necessary to address the court's concerns. The 1988 rule was also challenged in court, but this time the D.C. Circuit upheld the rule. *Union of Concerned Scientists v. U.S. Nuclear Regulatory Com'n*, 880 F.2d 552 (1989).

In its current form, 10 CFR 50.109(a)(1) defines backfitting as ~~"~~

> ... the modification of or addition to systems, structures, components, or design of a facility; or the design approval or manufacturing license for a facility; or the procedures or organization required to design, construct or operate a facility; any of which may result from a new or amended provision in the Commission's regulations or the imposition of a regulatory staff position interpreting the Commission's regulations that is either new or different from a previously applicable staff position ....

~~. . . ."~~ Unless one of three specified exceptions apply, the NRC may impose a backfit only if it performs a backfit analysis in accordance with 10 CFR 50.109(a)(2) and determines in accordance with 10 CFR 50.109(a)(3) "that there is a substantial increase in the overall protection of the public health and safety or the common defense and security to be derived from the backfit and that the direct and indirect costs of implementation for that facility are justified in view of this increased protection."

Section 50.109(a)(4) sets forth the three exceptions to the requirements of 10 CFR 50.109(a)(2) and (a)(3). The first exception, the compliance exception, applies if the "modification is necessary to bring a facility into compliance with a license or the rules or orders of the Commission, or into conformance with written commitments by the licensee." 10 CFR 50.109(a)(4)(i). The second and third exceptions relate to actions ensuring adequate protection or to actions that involve defining or redefining adequate protection. 10 CFR 50.109(a)(4)(ii)-(iii).

Commission Policy

The Commission addressed its intended application of the compliance exception in the 1985 rulemaking (50 FR at 38103):

> The compliance exception is intended to address situations in which the licensee has failed to meet known and established standards of the Commission because

of omission or mistake of fact. It should be noted that new or modified interpretations of what constitutes compliance would not fall within the exception and would require a backfit analysis and application of the standard.

In the 1985 rule, the Commission acknowledged that staff interpretations of regulations are not legally binding, but the Commission also stated that "staff interpretations of broadly stated rules are often necessary to give a rule effect and in some instances may be a causal factor in initiating a backfit." *Id.* at 38102. The Commission also stated, "Many of the most important changes in plant design, construction, operation, organization, and training have been put in place at a level of detail that is expressed in staff guidance documents which interpret the intent of broad, generally worked [sic] regulations." *Id.* at 38103.[4]

Backfitting Guidance

Extensive information regarding the appropriate implementation of backfitting is provided in the NRC's 1990 Backfitting Guidelines (NUREG-1409). Relevant excerpts from this guidance are provided below.

Applicable Regulatory Staff Positions

According to NUREG-1409, t~~T~~o be a backfit, "a new or revised staff position or requirement must be involved, that is, there must be a change in content or applicability of the previously applicable regulatory staff position (in the direction of increased safety requirements) . . . ." ~~(at 3)~~ An applicable regulatory staff position is a requirement or position already specifically imposed on or committed to by a licensee. Examples of applicable regulatory staff positions include:

- ~~A requirement or position already specifically imposed on or committed to by a licensee is called an applicable regulatory staff position. There are several different types of positions, such as~~

- legal requirements, as in explicit regulations, orders, and plant licenses and in amendments, conditions, and technical specifications

- written licensee commitments such as those contained in the final safety analysis report, licensee event reports, and docketed correspondence, including responses to NRC bulletins, generic letters, inspection reports, or notices of violation and confirmatory action letters

- NRC staff positions that are documented explicit interpretations of more general regulations and are contained in documents such as the Standard Review Plan, branch technical positions, regulatory guides, generic letters, and bulletins

~~For the purpose of this report, a change in the applicable regulatory staff position will be subsequently referred to as a new or revised position.~~

~~This manual chapter is included as~~A similar list of examples is provided in Inspection Manual Chapter 0514 (1988), which is included as Appendix D to NUREG-1409. The manual chapter was referenced in the 1988 rulemaking, and a working draft was provided to the Commission in

[4] The 1988 rulemaking neither revised the compliance exception as stated in the 1985 rule nor provided additional guidance on its interpretation.

SECY-88-102 for information. The manual chapter provides a definition of "applicable regulatory staff positions" that is slightly more detailed than the definition in NUREG-1409. This definition is quoted below, with additional detail beyond the NUREG-1409 emphasized in underlined text.

> Applicable regulatory staff positions are those already specifically imposed upon or committed to by a licensee at the time of the identification of a plant-specific backfit, and are of several different types and sources:
>
> a. Legal requirements such as in explicit regulations, orders, plant licenses (amendments, conditions, technical specifications). Note that some regulations have update features built in, as for example, 10 CFR 50.55a, Codes and Standards. Such update requirements are applicable as described in the regulation.
>
> b. Written commitments such as contained in the [Final Safety Analysis Report], [Licensee Event Reports], and docketed correspondence, including responses to Bulletins, responses to Generic Letters, Confirmatory Action Letters, responses to Inspection Reports, or responses to Notices of Violation.
>
> c. NRC staff positions[5] that are documented, approved, explicit interpretations of the more general regulations, and are contained in documents such as the [Standard Review Plan], Branch Technical Positions, Regulatory Guides, Generic Letters, and Bulletins; and to which a licensee or an applicant has previously committed to or relied upon. Positions contained in these documents are not considered applicable staff positions to the extent that staff has, in a previous licensing or inspection action, tacitly or explicitly excepted the licensee from part or all of the position.[6]

[5] Requirements may be imposed by rule or order. Staff interpretations such as examples of acceptable ways to meet requirements are not requirements in and of themselves.
[6] Imposition of a staff position from which a licensee has previously been excepted is a backfit.

How Regulatory Positions are Established

NUREG-1409 provides responses to a number of questions regarding backfitting. The following response was given to questions asking, "Is it appropriate for the NRC staff to rely on informal or formal communications to other licensees as official NRC positions? What about NRC tacit approval of documents?"

> Informal or formal communications to one licensee are not official positions to all licensees. Section 053 of Manual Chapter 0514 identifies what can be applied as official staff positions in a plant-specific context. They are legal requirements such as contained in explicit regulations, orders, and plant licenses; written commitments such as contained in final safety analysis reports, licenses event reports, and docketed correspondence; and documented, approved explicit interpretations such as contained in the [Standard Review Plan], branch technical positions, regulatory guides, generic letters, and bulletins. Orders, licenses, and written commitments are applicable only to a particular licensee.
>
> If the NRC staff previously exempted a licensee from a legal requirement or approved position, it is not applicable to that licensee for the purpose of backfit consideration. Explicit exemption would be done formally in writing. The Appendix to NRC Manual Chapter 0514 discusses tacit approval under reanalysis of issues. Two situations are covered. In the first case, staff review of a previously accepted licensee action or program may result in a requested change. This would be classified as a backfit because it represents a change in a previous staff position and would require a backfit analysis (or a documented evaluation if it meets one of the exceptions listed in the backfit rule). In the second case, a licensee submittal committing to a specific course of action that has not received timely NRC staff review is implemented by the licensee. In this case, it is considered that the NRC staff tacitly accepted the licensee's action since timely notice to the contrary was not given. If the NRC staff subsequently adopts a different position and requests a change in the licensee action, this change may be classified as a backfit and thus require a backfit analysis (or a documented evaluation if it meets one of the exceptions listed in the backfit rule).

NUREG-1409 also addresses a question regarding tacit approvals by an inspector: "If an inspector has previously accepted (i.e., provided tacit approval of) a licensee's method, does a specific request for change constitute a backfit and if so, is a backfit analysis required?" The response is:

> Cases where an inspector provides tacit approval are relatively rare. Simply not challenging a licensee's practice normally would not be considered tacit approval. The only example provided in Manual Chapter 0514 is a case where the NRC has indicated tacit approval by not acting in a reasonable time on a licensee submittal and the licensee has moved ahead to implement the proposal described in the submittal. For the purpose of this question, it would most likely arise in connection with review of a licensee response to an inspection report.
>
> Explicit approval could be provided in an inspection report that states that a particular approach is acceptable. However, conclusions of that nature are usually made in SEs rather than inspection reports.

Compliance Backfit Guidance

~~The Backfitting Guidelines~~NUREG-1409 gives the following response to ~~include~~ the question, ~~(on page 12)~~ "[h]ow does the backfit rule apply to new staff positions that reflect an evolving understanding of technical issues?" ~~The response is:~~

> ~~New or revised staff positions are backfits when they are imposed on licensees and result in a change in structures, systems, design, or procedures (as described in 10 CFR 50.109). A backfit analysis is required whenever new or revised positions are imposed to achieve cost-justified substantial safety enhancements. A backfit analysis is not required if the new or changed position is imposed to bring a facility into compliance or if it is necessary to provide assurance of adequate protection. In those cases, however, a written evaluation is needed to provide the objectives of and reasons for the modification and the basis for invoking the exception.~~
>
> An evolving understanding of issues does not, by itself, define which category fits a particular backfit. Judgment must be applied to the facts of each particular case to determine whether the backfit is for compliance, to provide adequate protection, to redefine adequate protection, or to achieve a cost-justified substantial safety enhancement. For example, with regard to compliance, the 1985 statement of considerations for 10 CFR 50.109 indicates that "the compliance exception is intended to address situations where the licensee has failed to meet known and established standards of the Commission because of omission or mistake of fact....new or modified interpretations of what constitutes compliance would not fall within the exception...."

NUREG-1409 also provides an example where an evolving understanding of technical issues resulted in a compliance backfit that was apparently justified for at least some licensees. ~~The Backfitting Guidelines further ask (on page 13) if it is "appropriate for the NRC staff to rely on informal or formal communications to other licensees as official NRC positions? What about NRC tacit approval of documents?" The response is:~~

> ~~Informal or formal communications to one licensee are not official positions to all licensees. Section 053 of Manual Chapter 0514 identifies what can be applied as official staff positions in a plant-specific context. They are legal requirements such as contained in explicit regulations, orders, and plant licenses; written commitments such as contained in final safety analysis reports, licenses event reports, and docketed correspondence; and documented, approved explicit interpretations such as contained in the [Standard Review Plan], branch technical positions, regulatory guides, generic letters, and bulletins. Orders, licenses, and written commitments are applicable only to a particular licensee.~~
>
> ~~If the NRC staff previously exempted a licensee from a legal requirement or approved position, it is not applicable to that licensee for the purpose of backfit consideration. Explicit exemption would be done formally in writing. The Appendix to NRC Manual Chapter 0514 discusses tacit approval under reanalysis of issues. Two situations are covered. In the first case, staff review of a previously accepted licensee action or program may result in a requested change. This would be classified as a backfit because it represents a change in a previous staff position and would require a backfit analysis (or a documented evaluation if it meets one of the exceptions listed in the backfit rule). In the~~

> ~~second case, a licensee submittal committing to a specific course of action that has not received timely NRC staff review is implemented by the licensee. In this case, it is considered that the NRC staff tacitly accepted the licensee's action since timely notice to the contrary was not given. If the NRC staff subsequently adopts a different position and requests a change in the licensee action, this change may be classified as a backfit and thus require a backfit analysis (or a documented evaluation if it meets one of the exceptions listed in the backfit rule).~~

~~The Backfitting Guidelines also consider an example in which the~~In response to industry claims that Bulletin 88-11 ~~lacks~~ lacked any backfitting justification, the staff responded~~. The response is~~:

> Although the justification was not printed in the bulletin, NRC Bulletin 88-11, "Pressurizer Surge Line Thermal Stratification," was justified as a backfit. It is an example of a backfit that was determined by the responsible NRC official to be required as a matter of compliance with existing requirements and commitments. The CRGR reviewed the bulletin and concurred. The regulations currently require licensees to meet the applicable codes of the American Society of Mechanical Engineers (ASME), *Boiler and Pressure Vessel Code*. Because of the NRC staff's concern with the integrity of the surge line, licensees were requested to perform their fatigue analysis in accordance with the latest ASME Section III requirements that incorporate high cycle fatigue analysis. The justification provided by the NRC staff was that previously unconsidered thermal stratification phenomenon may invalidate the existing analysis performed to confirm the integrity of the surge line.
>
> Subsequently, it was understood that some licensees believed that the NRC staff's rationale was in error because they were not committed to the latest ASME Section III requirements by virtue of their license commitment. However, the issue became moot because these licensees undertook the analysis voluntarily in view of the safety importance of the issue and the fact that previous versions of the ASME Code did not completely address the concern.

~~Finally, the Backfitting Guidelines (on page 15) pose the question that "[i]f an inspector has previously accepted (i.e., provided tacit approval of) a licensee's method, does a specific request for change constitute a backfit and if so, is a backfit analysis required?" The response is:~~

> ~~Cases where an inspector provides tacit approval are relatively rare. Simply not challenging a licensee's practice normally would not be considered tacit approval. The only example provided in Manual Chapter 0514 is a case where the NRC has indicated tacit approval by not acting in a reasonable time on a licensee submittal and the licensee has moved ahead to implement the proposal described in the submittal. For the purpose of this question, it would most likely arise in connection with review of a licensee response to an inspection report.~~
>
> ~~Explicit approval could be provided in an inspection report that states that a particular approach is acceptable. However, conclusions of that nature are usually made in SEs rather than inspection reports.~~

NRC ~~Manual Chapter 0514 (1988)~~

This manual chapter is included as Appendix D to NUREG-1409. The manual chapter was referenced in the 1988 rulemaking, and a working draft was provided to the Commission in SECY-88-102 for information. The manual chapter provides a definition of "applicable regulatory staff positions" that is slightly more detailed than the definition in NUREG-1409. This definition is quoted below, with additional detail beyond the NUREG-1409 emphasized in underlined text.

> Applicable regulatory staff positions are those already specifically imposed upon or committed to by a licensee at the time of the identification of a plant-specific backfit, and are of several different types and sources:
>
> a. Legal requirements such as in explicit regulations, orders, plant licenses (amendments, conditions, technical specifications). Note that some regulations have update features built in, as for example, 10 CFR 50.55a, Codes and Standards. Such update requirements are applicable as described in the regulation.
>
> b. Written commitments such as contained in the [Final Safety Analysis Report], [Licensee Event Reports], and docketed correspondence, including responses to Bulletins, responses to Generic Letters, Confirmatory Action Letters, responses to Inspection Reports, or responses to Notices of Violation.
>
> c. NRC staff positions[7] that are documented, approved, explicit interpretations of the more general regulations, and are contained in documents such as the [Standard Review Plan], Branch Technical Positions, Regulatory Guides, Generic Letters, and Bulletins; and to which a licensee or an applicant has previously committed to or relied upon. Positions contained in these documents are not considered applicable staff positions to the extent that staff has, in a previous licensing or inspection action, tacitly or explicitly excepted the licensee from part or all of the position.[8]

~~[7] Requirements may be imposed by rule or order. Staff interpretations such as examples of acceptable ways to meet requirements are not requirements in and of themselves.~~

~~[8] Imposition of a staff position from which a licensee has previously been excepted is a backfit.~~

APPENDIX B: QUALIFICATION OF PRESSURE RELIEF VALVES IN NUCLEAR POWER PLANTS IN RESPONSE TO TMI-2 ACCIDENT

Nuclear power plants in the United States use various types of pressure relief valves to protect personnel and equipment from overpressure events within reactor fluid systems. Pressure relief valves include safety valves, safety relief valves, and relief valves, with different designs, operating conditions, and requirements. The American Society of Mechanical Engineers (ASME) *Boiler and Pressure Vessel Code* (BPV Code), Section III, Division 1, specifies requirements for the design, operation, installation, and testing of pressure relief valves used for various functions in nuclear power plants. For example, the ASME BPV Code (2007 Edition) in Article NB-7000, Overpressure Protection, specifies requirements for steam and air or gas service for safety valves; steam, air or gas, and liquid service for safety relief valves; liquid service for relief valves; and steam, air or gas, and liquid service for pilot operated or power actuated pressure relief valves. The ASME *Code for Operation and Maintenance of Nuclear Power Plants* (OM Code) provides requirements for the preservice and inservice testing (IST) programs for pressure relief valves in nuclear power plants.

Braidwood, Units 1 and 2 (Braidwood) and Byron, Units 1 and 2 (Byron) are Westinghouse-designed pressurized-water reactors (PWRs) that received their construction permits under Title 10 of the *Code of Federal Regulations* (10 CFR), Part 50, in December 1975. Each pressurizer in these four reactor units is equipped with three pressurizer safety valves (PSVs) and two power-operated relief valves (PORVs) The three PSVs are Crosby Model HP-BP-86, size 6M6 (6-inch), spring loaded pop type opened by direct fluid pressure. The PORVs are Copes-Vulcan Model D-100-160 3-inch pneumatic-actuated globe valves that respond to a signal from the pressure sensing system or to manual control. Each PORV can be isolated by a motor-operated block valve.

The ASME BPV Code of record for the PSVs at Braidwood and Byron was the 1971 Edition through the Winter 1972 addenda of the ASME BPV Code, Section III. At the time of the Braidwood and Byron operating license review, NRC Standard Review Plan (SRP), Revision 1 (July 1981), Chapter 15.5.1-15.5.2, "Inadvertent Operation of ECCS and Chemical and Volume Control System Malfunction that Increases Reactor Coolant Inventory," and Chapter 15.6.1, "Inadvertent Opening of a PWR Pressurizer Pressure Relief Valve or a BWR [boiling-water reactor] Pressure Relief Valve," provided general staff guidance for these plant transients. In March 2007, the NRC staff issued Revision 2 to these SRP chapters with significantly more detail, including a statement that PSVs and PORVs are assumed to fail open if they relieve water without being qualified.

The accident at Three Mile Island, Unit 2 (TMI-2) on March 28, 1979, included failure of a PORV on the pressurizer to reclose properly during the event. Based on lessons learned from the TMI-2 accident, the NRC issued recommendations regarding performance testing of safety and relief valves used in nuclear power plants in NUREG-0578 (July 1979), "TMI-2 Lessons Learned Task Force Status Report and Short-Term Recommendations." In particular, the NRC staff recommended in Section 2.1.2, "Performance Testing for BWR and PWR Relief and Safety Valves," of NUREG-0578 that nuclear power plant licensees commit to provide performance verification by full-scale prototypical testing for all relief and safety valves.

On October 31, 1980, the NRC issued a letter to all then-operating nuclear power plants and applicants for operating licenses and holders of construction permits forwarding NUREG-0737, "Clarification of TMI Action Plan Requirements." Requirement II.D.1, "Performance Testing of

Boiling-Water Reactor and Pressurized-Water Reactor Relief and Safety Valves (NUREG-0578, Section 2.1.2)," in NUREG-0737 specified the NRC position that PWR and BWR licensees and applicants shall conduct testing to "qualify" the reactor coolant system (RCS) relief and safety valves under expected operating conditions for design-basis transients and accidents. The detailed clarification in NUREG-0737 of this NRC position specified the following:

> Licensees and applicants shall determine the expected valve operating conditions through the use of analyses of accidents and anticipated operational occurrences referenced in Regulatory Guide 1.70, Revision 2. The single failures applied to these analyses shall be chosen so that the dynamic forces on the safety and relief valves are maximized. Test-pressures shall be the highest predicted by conventional safety analysis procedures. Reactor coolant system relief and safety valve qualification shall include qualification of associated control circuitry, piping, and supports, as well as the valves themselves.
>
> A. Performance Testing of Relief and Safety Valves--The following information must be provided in report form by October 1, 1981:
>
> (1) Evidence supported by test of safety and relief valve functionability for expected operating and accident (non-ATWS) conditions must be provided to NRC. The testing should demonstrate that the valves will open and reclose under the expected flow conditions.
>
> (2) Since it is not planned to test all valves on all plants, each licensee must submit to NRC a correlation or other evidence to substantiate that the valves tested in the EPRI (Electric Power Research Institute) or other generic test program demonstrate the functionability of as-installed primary relief and safety valves. This correlation must show that the test conditions used are equivalent to expected operating and accident conditions as prescribed in the final safety analysis report (FSAR). The effect of as-built relief and safety valve discharge piping on valve operability must also be accounted for, if it is different from the generic test loop piping.
>
> (3) Test data including criteria for success and failure of valves tested must be provided for NRC staff review and evaluation. These test data should include data that would permit plant-specific evaluation of discharge piping and supports that are not directly tested.

In describing the type of review to be conducted for this regulatory position, the NRC staff stated the following:

> Pre-implementation review will be performed for EPRI and BWR test programs with respect to qualification of relief and safety valves. Also, the applicants' proposal for functional testing or qualification of PWR valves will be reviewed. Post-implementation review will also be performed of the test data and test results as applied to plant-specific situations.

In specifying the documentation required to satisfy this regulatory position, the NRC staff stated the following:

> Pre-implementation review will be based on EPRI, BWR, and applicant submittals with regard to the various test programs. These submittals should be

made on a timely basis as noted below, to allow for adequate review and to ensure that the following valve qualification dates can be met:

Final PWR (EPRI) Test Program--July 1, 1980

Final BWR Test Program--October 1, 1980

Block Valve Qualification Program--January 1, 1981

Post-implementation review will be based on the applicants' plant-specific submittals for qualification of safety relief valves and block valves. To properly evaluate these plant-specific applications, the test data and results of the various programs will also be required by the following dates:

PWR (EPRI)/BWR Generic Test Program Results--July 1, 1981

Plant-specific submittals confirming adequacy of safety and relief valves based on licensee/applicant preliminary review of generic test program results--July 1, 1981

Plant-specific reports for safety and relief valve qualification--October 1, 1981

Plant-specific submittals for piping and support evaluations--January 1, 1982

Plant-specific submittals for block valve qualification--July 1, 1982.

In a letter dated July 27, 1982, to the NRC staff, the Westinghouse Owners Group (WOG) submitted WCAP-10105 (June 1982), "Review of Pressurizer Safety Valve Performance as Observed in the EPRI Safety and Relief Valve Test Program." In WCAP-10105, the WOG indicated that the design specification for PSVs in Westinghouse-designed nuclear power plants is for steam service only. Based on a review of the EPRI test data, the WOG concluded that the valves performed with chatter, but did not identify any valve damage. (ADAMS LL Accession No. 8208190310, Microfiche 14387:191-301)

In December 1982, EPRI issued NP-2628-SR, "EPRI PWR Safety and Relief Valve Test Program – Safety and Relief Valve Test Report," that described safety and relief valve tests for types of valves in service at nuclear power plants. In particular, Section 3.5 in EPRI NP-2628-NP discusses the testing of Crosby safety valves similar to the PSVs at Braidwood and Byron, including two water tests. The report indicated chattering of the safety valves with subsequent inspection finding galled surfaces and damage to internal parts. Section 4.6 in EPRI NP-2628 discussed testing of Copes-Vulcan relief valves similar to the pressurizer PORVs at Braidwood and Byron, although the extent of water testing is not fully described. The report indicated no damage found during the inspection of the Copes-Vulcan relief valves. The report did not indicate any failures of the Crosby or Copes-Vulcan valves to reseat during the testing. (ADAMS LL Accession No. 8407130197, Microfiche 25588:082-262)

In January 1983, EPRI issued NP-2770-LD, "EPRI/C-E PWR Safety Valve Test Report," that described the testing of PWR primary system safety valves. Volume 1 provides a summary of the test program and its results. Section 4.5 of Volume 1 indicates that the following tests were performed on the Crosby 6M6 PSV: 11 steam tests with filled loop seals, 3 steam-to-water transition tests, and 2 water tests. The report states that the valve experienced chatter during the tests, and one water test had to be terminated. The individual volumes of EPRI NP-2770-LD discuss the test results for each specific PSV type. Volume 6 provides the test details for the

Crosby 6M6 PSV. (EPRI NP-2770-LD, Volume 1, was obtained as a public document from the EPRI website. EPRI NP-2770-LD, Volume 6, could not be located within ADAMS or the NRC Record Retention Files, but is available for a fee from EPRI.)

In October 1982, EPRI issued NP-2670-LD, "EPRI/Wyle Power-Operated Relief Valve Phase III Test Report," to address testing of PORVs. This document could not be located in ADAMS despite its reference by nuclear power plant licensees. See, for example, North Anna Units 1 and 2 UFSAR (Revision 51, dated September 30, 2015), Section 15.2.14, "Spurious Operation of the Safety Injection System at Power."

The NRC review of the operating license applications for Braidwood and Byron included evaluation of the TMI ~~action~~ Action Plan items as discussed in the NRC Safety Evaluation Report (SER) for Braidwood Units 1 and 2, NUREG-1002, Section 1.1, "Introduction." In this SER section, the NRC staff stated that the review and evaluation of compliance by the applicant with the licensing requirements established in NUREG-0660, "NRC Action Plan Developed as a Result of the TMI-2 Accident," and NUREG-0737 (including item II.D.1 in Table 1.1) were incorporated into the reviews summarized throughout the SER. The NRC SER for Byron Units 1 and 2, NUREG-0876, also includes discussions of the NRC staff review of the TMI ~~action~~ Action Plan items.

Appendix E, "Requirements Resulting from TMI-2 Accident," to the Braidwood/Byron UFSAR in Section E.23, "Relief and Safety Valve Test Requirements (II.D.1)," indicated that a letter dated April 1, 1982, from D. Hoffman (Consumers Power) transmitted the Safety and Relief Valve Test Report for the EPRI PWR Safety and Relief Valve Test Program. The UFSAR stated that the final evaluation of the data indicated that the relief and safety valves will perform their intended functions for all expected fluid inlet conditions. The UFSAR also indicated that the plant-specific final evaluation confirming the adequacy of the relief and safety valves had been submitted by a letter from T. Tramm, dated October 26, 1982.

In Supplement 1 to the Braidwood SER (NUREG-1002, Supplement 1, September 1986), in Section 3.9.3.3, "Design and Installation of Pressure Relief Devices," the NRC staff stated that EPRI had completed a full-scale valve testing program, and that the WOG submitted the test results in WCAP-10105 in a letter dated July 27, 1982, from O. Kinglsey to S. Chilk. (ADAMS LL Accession No. 8208190307, Microfiche 14387:189-301) The NRC staff stated that the applicant responded to a requirement to demonstrate operability of these valves through submittals dated July 1 and October 26, 1982, and December 30, 1983. On the basis of a preliminary review, the NRC staff concluded that the applicant's general approach to responding to this item was acceptable, and provided adequate assurance that the RCS overpressure protection systems at Braidwood can adequately perform their intended functions. The NRC staff stated that if the detailed review revealed modifications or adjustments to safety valves, PORVs, PORV block valves, or associated piping, were needed to ensure that all intended design margins were present, the NRC staff would require that the applicant make appropriate modifications. The NRC staff categorized this issue as a Confirmatory Item. In Supplement 5 to the Byron SER (NUREG-0876, Supplement 5, October 1984) in Section 3.9.3.3, the NRC staff provided a similar discussion of the status of the NRC review of the capability of the Byron pressurizer valves. In Supplement 8 to the Byron SER (March 1987), the NRC staff stated TMI Action Plan Item II.D.I (3.9.3.3) had been closed in Supplement 5 to the Byron SER. The NRC issued operating licenses for Byron Unit 1 in February 1985 and Unit 2 in January 1987, and Braidwood Unit 1 in July 1987 and Unit 2 in May 1988.

Following the issuance of the Byron and Braidwood operating licenses, the NRC staff provided a letter dated August 18, 1988, from L. Olshan to H. Bliss, indicating that Idaho National Engineering Laboratory (INEL) Technical Evaluation Report (TER) EGG-NTA-8028 (January 1988) provided the review of the Byron response to NUREG-0737, Item II.D.1. (ADAMS LL Accession No. 8808260355, Microfiche 46653:240-269) The NRC staff indicated that the licensee should develop and adopt plant procedures to inspect the pressurizer valves after each lift involving loop seal or water discharge. The TER described the INEL review of the EPRI testing of a PSV and PORV similar to the Byron pressurizer valves. The TER indicated that the PSV had two applicable tests: a loop seal/steam water transition test where the valve opened, chattered and stabilized to close; and a saturated water test where the valve opened with water, chattered, and stabilized. The TER indicated that the PORV opened and closed on demand in the loop seal/steam water transition test with a bending moment that was evaluated by analysis. The TER concluded that Byron provided an acceptable response to NUREG-0737, Item II.D.1. On May 21, 1990, the NRC staff provided a letter from S. Sands to T. Kovach with the Braidwood TER that included similar findings. (ADAMS LL Accession No. 9005290209, Microfiche 53927:301-330)

In January 1988, WCAP-11677, "Pressurizer Safety Relief Valve for Water Discharge During a Feedwater Line Break," provided a description of the WOG comparison of the EPRI test data with feedline break safety analyses. This report was submitted as an attachment to a response to a request for additional information (RAI) dated May 8, 1989, from the licensee of the Seabrook nuclear power plant. (ADAMS Microfiche 49775:336 – 49756:017) As discussed in the report, the WOG determined that all nuclear power plants addressed in the EPRI testing have PSVs that will operate reliably during water relief. The WOG evaluated the performance of the Crosby 6M6 PSVs during the EPRI tests, and considered that the performance involved less significant flutter (half lift motion) than the chatter (full lift motion) determined in the EPRI report. The WOG concluded that the Crosby 6M6 PSV can pass slightly subcooled water at a minimum up to three times without damage.

APPENDIX C: CONCERNS REGARDING PERFORMANCE OF PRESSURIZER VALVES UNDER WATER FLOW CONDITIONS

Westinghouse Nuclear Safety Advisory Letter

In 1993 and 1994, Westinghouse issued Nuclear Safety Advisory Letter (NSAL) 93-013 (June 30, 1993) and NSAL-93-013, Supplement 1 (October 28, 1994) to operating nuclear power plants (including Braidwood and Byron). These advisories resulted from Westinghouse's discovery that potentially nonconservative assumptions were used in the licensing analysis of the Inadverent Operation of the Emergency Core Cooling System at Power (IOECCS) event.

In NSAL-93-013, Westinghouse recommended that licensees determine if their pressurizer safety relief valves (PSRVs) are capable of closing following discharge of subcooled water. Westinghouse noted that the PSRVs might have been designed or "qualified" to relieve subcooled water. Westinghouse indicated that water relief through the power-operated relief valves (PORVs) is not a concern, because the PORV block valves can be used to isolate the PORVs if they fail to close. If the PSRVs are not designed or qualified for subcooled water relief, Westinghouse recommended that licensees re-evaluate the IOECCS event with three possible options of (1) reducing ECCS flow used in the safety analysis, (2) using a less restrictive operator response time, or (3) crediting the use of one or more PORVs to help mitigate the event.

In Supplement 1 to NSAL-93-013, Westinghouse ~~alerted~~ informed licensees ~~to~~ of a potential reduced time for operator action if a positive displacement pump is in service, and to the need to qualify the PSRVs and the piping downstream of the PSRVs and PORVs if water relief from the pressurizer is predicted.

Some licensees of operating nuclear power plants ~~alerted~~ informed the NRC ~~to~~ of their actions to address the potential concerns regarding liquid service for pressurizer safety valves (PSVs) and PORVs. A sample of actions by nuclear power plant licensees is summarized below in the "Plant-Specific Actions" section.

Additional NRC Generic Communications and Guidance

In December 2003, the NRC staff issued NRR Review Standard for Extended Power Uprates (RS-001, Rev. 0)~~—~~. Item 8 on page 7 of the review standard states that pressurizer level should not be allowed to reach a pressurizer water-solid condition.

On December 14, 2005, the NRC issued Regulatory Issue Summary (RIS) 2005-~~029~~-29, "Anticipated Transients that could Develop into More Serious Events," to notify nuclear power plant licensees of a concern identified during recent reviews of power uprate LARs. In RIS 2005-~~029~~-29, the NRC staff stated that typically Condition II event scenarios involve discharging water through relief or safety valves that are not qualified for water relief. The NRC staff stated that these valves are then assumed to fail in the open position and create a small break LOCA. The NRC staff stated that it was concerned that some licensees may be crediting PORVs without qualification for water relief and without establishing additional restrictions to ensure the availability of PORVs and block valves. The NRC staff stated that Westinghouse NSAL-93-013 allowing block valves to isolate PORVs is inconsistent with non-escalation criterion.

In proposed Revision 1 to RIS 2005-29, the NRC staff addresses the specific Condition II scenarios of chemical volume and control system (CVCS) malfunction, IOECCS event, and inadvertent opening of a PORV or PSV. Regarding the CVCS malfunction, the NRC staff states that performing only the reactivity anomaly analysis or assuming that this malfunction is not as severe as the IOECCS event is not acceptable. Regarding the IOECCS event, the NRC staff states that five of the alternative approaches in ~~NSAL-93~~NSAL-93-013 fail to meet the non-escalation criterion. The NRC staff indicated that these unacceptable alternative approaches are (1) closing the block valve, (2) assuming that the PORV is not operable, (3) addressing a stuck-open PORV or PSV as a separate Condition II event, (4) determining that a stuck-open PORV or PSV is not as severe as a small break LOCA, and (5) determining that RCS loss through PORV is made up by ECCS flow. Regarding inadvertent opening of PORV or PSV, the NRC staff states that inadvertent opening of PSV or PORV could continue as a Condition III small break LOCA and fails to meet the non-escalation criterion.

Additional General PSV/PORV Information

In August 2004, EPRI issued Report 1011047, "Probability of Safety Valve Failure-to-Reseat Following Steam and Liquid Relief - Quantitative Expert Elicitation," 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 relief through safety valves might cause increased chatter, and therefore, an increased failure rate.

In March 2011, the NRC published NUREG/CR-7037, "Industry Performance of Relief Valves at U.S. Commercial Nuclear Power Plants through 2007," based on a study by the Idaho National Laboratory. With respect to pressurizer PORVs, the report found four separate liquid relief events at four PWR plants. The report estimated 698 total demands on these PORVs during their liquid relief events with no failures to close. The report also summarized test data from EPIX for three valve types. The report indicates 2 failures of PORVs to reclose during 2070 demands, but does not specify liquid or steam service for the EPIX test information. With respect to PSVs, the report indicates 2 failures out of 4 total demands following plant scrams, but does not indicate liquid or steam service. NRC staff from the Office of Nuclear Regulatory Research provided Licensee Event Report information indicating that the 2 PSV failures involved reseating of the valves with leakage of 25 and 200 gpm, respectively. The report summarized EPIX test data for PSVs as no failures to reclose during 1805 demands.

Plant-Specific Actions

Diablo Canyon

On August 13, 1996, the licensee of the Diablo Canyon nuclear power plant submitted a report under 10 CFR 50.59 related to the potential for an IOECCS event. (ADAMS Microfiche 89419:294-322) The submittal included NSAL-93-013 and its supplement as enclosures. The licensee indicated that the PSVs had not been initially qualified for water relief, but were subsequently qualified for a brief period. The licensee indicated that WCAP-11677 was applicable and demonstrated that the PSVs were operable.

On July 2, 2004, the NRC granted a license amendment request (LAR) for Diablo Canyon that allowed credit for actuation of the PORVs in response to inadvertent safety injection (SI) actuation to avoid challenges to the PSVs. (ADAMS Accession No. ML041950300) In support of that LAR, the licensee responded on November 21, 2003, to requests for additional information (RAIs) related to the capability of the PORVs to function adequately under conditions predicted for design-basis transients and accidents. (ADAMS Accession No. ML033360735) In response to an RAI regarding the design adequacy of the PORVs if the pressurizer becomes water solid, the licensee had stated that the NRC had issued a letter dated January 26, 1986, "Safety and Relief Valve Testing, NUREG-0737 Item II.D.1," that provided an SER that accepted the adequacy of the PORV and block valve design and confirmatory testing for a range of fluid conditions (full pressure steam, steam to water transition, and subcooled water fluid).

Salem

On June 4, 1997, the NRC granted a technical specification (TS) revision for the Salem nuclear power plant to ensure that the automatic capability of the PORVs to relieve pressure is maintained. (ADAMS Accession No. ML011720397) In response to ~~NSAL-93~~NSAL-93-013, the licensee determined that an inadvertent ~~safety injection (SI)~~ actuation at power could cause the pressurizer to become water solid and PSVs lifting with water relief if the automatic operation of the PORVs is not made available for reactor coolant system (RCS) depressurization early in the transient. In that the Salem PSVs were not designed to relieve water, it was noted that water relief has the potential to cause the PSVs to fail in the open position.

In the course of the review of the licensee's application, the NRC staff noted that the PORVs were not designed to "safety related" standards and, thus, could not be credited for mitigation of the inadvertent SI actuation at power incident when the PORV is operating in the automatic mode. In response, the licensee proposed an upgrade of the PORVs to eliminate the possibility that a single active failure of a PORV component could prevent the mitigation of the inadvertent SI actuation at power incident. As discussed in the SER, the licensee implemented modifications to the PORV circuitry to qualify the upgraded circuitry as safety-related.

Regarding PORV performance, the licensee evaluated the PORV air accumulators for sufficient capacity for the inadvertent SI event. The licensee also reported that endurance tests had been performed with five different trims (with different trim materials) on one PORV at Wyle Laboratories to demonstrate that (1) after 2000 consecutive operations, there were no packing leaks nor packing gland adjustments required; (2) there was no diaphragm failure; and (3) the solenoid valve withstood 10,000 operations without any loss of function. Based on this information, the NRC staff concluded that the PORV performance was acceptable regarding the mitigation of an inadvertent SI event.

Millstone, Unit 3

On June 5, 1998, the NRC granted a license amendment for Millstone, Unit 3 for a TS revision to ensure that the capability of the PORVs to relieve pressure is maintained. (ADAMS Accession No. ML011800207) The revised TS Bases stated that the PORVs and their associated piping have been demonstrated to be "qualified" for water relief. The PORVs prevent water relief from the PSVs for which qualification for water relief has not been demonstrated. The TS Bases also stated that the prime importance for the capability to close the block valve is to isolate a stuck-open PORV. In the SER, the NRC staff stated that the licensee notified the NRC of the issue of potential water relief through the PSVs that could lead to valve failure in LER 97-063-00 on December 31, 1997.

To provide added assurance that the PSVs will not be damaged due to water relief during an ISI event, the licensee upgraded the PORV circuitry, added additional PORV surveillance requirements, qualified the PORVs and associated piping for water relief, and made emergency procedure changes to allow plant operators additional time to terminate the event. With respect to the PORV circuitry, the NRC staff concluded that the PORV circuitry modifications qualified the PORV control circuitry as safety-related. With respect to PORV performance, the licensee reanalyzed the inadvertent SI event with the LOFTRAN computer code to demonstrate that the PORVs were qualified for water relief for approximately 1 hour. The licensee referenced EPRI testing documented in NP-2670-LD, Volume 11, that was said to generically resolve post TMI-2 issues associated with PORVs and safety valve qualification for water and steam relief, with the results from four tests of a Garrett PORV (such as used at Millstone, Unit 3) for water relief. The licensee determined that the PORVs and associated piping are qualified for 1 hour of water relief for an IOECCS event. The licensee also stated that the PORV manufacturer performed numerous cycle tests to verify the performance of the valve design, and also verified that valve seat leakage was acceptable. The licensee stated that the PORV block valves had been evaluated for water relief in accordance with the program established in response to Generic Letter (GL) 89-10, "Safety-Related Motor-Operated Valve Testing and Surveillance." The NRC staff found the licensee information regarding the qualification of the PORVs for water relief during the inadvertent SI event to be acceptable.

Callaway

On September 25, 2000, the NRC granted a license amendment for the Callaway nuclear power plant to revise the TS to change the PSV lift setting range. (ADAMS Accession No. ML003753326) To prevent water passing through the PSVs during an IOECCS event, the licensee modified and upgraded the PORV circuitry to full Class 1E to take credit for automatic action of at least one PORV during the event. These actions would prevent water relief through the PSVs. In its TS revision request dated May 25, 2000, the licensee had stated that the design function of the valves was not being changed and the conclusions documented in the NRC SER of Callaway's response to NUREG-0737 Item II.D.1 (dated September 10, 1987) are unchanged. As a result, the licensee stated that the PORVs and associated discharge piping can accommodate water relief.

Braidwood and Byron

On May 29, 1998, the Braidwood and Byron licensee proposed an amendment to its TS to take credit for the automatic operation of the PORV to provide mitigation for an IOECCS event. In the amendment request, the licensee stated that the PSVs have not been qualified to reseat after passing subcooled liquid. The licensee stated that the PORVs at Braidwood and Byron are safety-related components with safety-related actuators and accumulator tanks with PORV control circuits classified as safety-related. The licensee noted that some portions of the PORV circuitry are nonsafety-related with improvements implemented in response to GL 90-06, Resolution of Generic Issue 70, "Power-Operated Relief Valve and Block Valve Reliability" and Generic Issue 94, "Additional Low-Temperature Over Pressure Protection for Light-Water Reactors" Pursuant to 10 CFR 50.54(f)." The licensee stated that the PORV block valves are within the scope of the GL 89-10 program. In a letter dated May 13, 1999, the NRC staff provided an RAI regarding the reliance on the PORVs that documented the basis for its concerns that the PORV circuitry did not meet the single failure criterion. In response to these concerns, the licensee withdrew its TS amendment request in a letter dated July 16, 1999. No further action regarding this amendment request has been identified.

On July 5, 2000, the Braidwood and Byron licensee submitted a request for a power uprate for Braidwood and Byron to increase the maximum thermal power for each unit from 3411 megawatts thermal (MWt) to 3586.6 MWt (commonly referred as a stretch power uprate). In

RAIs, the NRC staff requested that the licensee address water solid conditions in the pressurizer because it had generally not accepted a solid pressurizer for an IOECCS event to order to avoid the potential for all three PSVs to be stuck open due to liquid relief through these safety valves. In its letter dated November 27, 2000, the licensee stated that Section 15.5.1, "Inadvertent Operation of Emergency Core Cooling System During Power Operation," of the UFSAR had been revised to credit the PSVs to pass water. The licensee discussed the EPRI testing program in response to NUREG-0737 with the results summarized in EPRI NP-2628-SR. The licensee referenced the NRC letters from L. Olshan to H. Bliss, dated August 18, 1988, and S. Sands to T. Kovach, dated May 21, 1990, transmitting the TERs with the results of the NRC's review of the Byron and Braidwood response to NUREG-0737, Item II.D.1, respectively.

On January 31, 2001, the Braidwood and Byron licensee provided a response to an RAI supplement from the NRC staff requesting the temperature of water to be passed by the pressurizer safeties and the length of time that the safeties are expected to pass water. The NRC staff also asked the licensee to discuss what EPRI tests are applicable to the Byron and Braidwood condition. In response, the licensee stated that the PSVs would close after passing water, although they may not be leaktight. The licensee stated that the leakage from up to three leaking PSVs is bounded by one fully open PSV. The licensee indicated that the EPRI testing of the Crosby safety valves in EPRI NP-2770-LD, Volumes 1 and 6, are applicable. The licensee indicated that valve chatter occurred during the tests with damage to the internals, but that the safety valve closed in response to system depressurization. The licensee stated that the Byron/Braidwood pressurizer water temperature of 590 °F is higher than the EPRI tests (530 °F). The licensee stated that the assumed length of the event is 20 minutes from initial SI signal to when the system pressure is restored below PSV lift setpoint.

In the NRC SER dated May 4, 2001, granting the Byron/Braidwood power uprate in Section 3.2, "Non-LOCA [loss-of-coolant accident] Transient Analysis," the NRC staff discussed its review of the performance of the PORVs and PSVs to discharge liquid water for approximately 20 minutes. (ADAMS Accession No. ML033040016) The NRC staff discussed the EPRI testing program with the conclusion that the safety valve closed in response to system depressurization. The NRC staff reviewed the licensee's evaluation of the performance of the PSVs for liquid water conditions. The NRC staff found that the EPRI tests adequately demonstrate the performance of the valves for the expected water temperature conditions and that there is reasonable assurance that the valves will adequately reseat following the spurious SI event. The NRC staff determined that a review of the 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. Therefore, the NRC staff found the licensee's crediting of the PSVs to discharge liquid water during the spurious SI event to be acceptable. This portion of the NRC SER was based on the specific review of PSV performance for the Byron and Braidwood power uprate request described in a memorandum dated March 15, 2001, from the NRR Reactor Systems Branch with technical input from the responsible staff member for safety valves in the NRR Division of Engineering (ADAMS Accession No. ML010740316).

As noted by the licensee, the Byron/Braidwood UFSAR at the time of the stretch power uprate (Revision 9, dated December 2002) in Chapter 15.5.1 includes PSV water relief, and references the INEL 1988 report and L. Olshan August 1988 SER. The current UFSAR Revision 15 (dated December 2014) concludes that the IOECCS event does not progress into a stuck-open PSV LOCA event. The UFSAR states that all three PSVs may lift but will reclose, and that the leakage is bounded by one fully open valve with the consequences bounded by the IOPSRV event.

On August 26, 2004, the NRC issued a license amendment for Braidwood and Byron granting an adjustment to the PSV setpoints. (ADAMS Accession No. ML042250531) In an RAI, the NRC staff requested that the licensee perform a quantitative analysis regarding PSV water cycles and relief/discharge water temperature. As for the loss of ac power (LOAC) with reactor coolant pump (RCP) seal injection event, the licensee's analysis indicated that continued injection of water into the RCS through the RCP seals would result in a water-solid pressurizer and water discharge through the PSVs. The proposed PSV setpoint tolerance assuming negative tolerance would result in a lower PSV lift setpoint. With the lower setpoint, the PSV would open earlier, and a larger number of PSV water cycles with a lower water discharge temperature could result during the transient. The licensee performed an analysis of the LOAC with RCP seal injection event, and determined the revised PSV setpoint would result in an increase of about one PSV water cycle and a reduction in the liquid discharge temperature of about 0.5 °F. A comparison of the reanalysis showed that the spurious SI event remained the limiting event since it resulted in a greater increase in the number of PSV water cycles (two cycles vs. one cycle) and a greater decrease in the PSV discharge water temperature (3.0 °F vs. 0.5 °F) than that calculated for the LOAC with RCP seal injection event. The water discharge temperature in the analysis of record for the spurious SI event was 590 °F. The lowest discharge water temperature for the spurious SI event with the revised PSV setpoint is 587 °F. The NRC staff found that the calculated water discharge temperature (587 °F) is significantly higher than the discharge water temperature of 530 °F that was used to support operability of the PSVs as discussed in the analysis of record. As a result, the NRC staff concluded that the reanalysis is acceptable to assure that the PSVs will remain operable following a spurious SI event.

On February 7, 2014, the NRC issued a license amendment for Braidwood and Byron granting a Measurement Uncertainty Recapture (MUR) power uprate. The NRC staff determined that the IOECCS event was outside of the scope of the MUR power uprate, because the licensee did not modify the Chapter 15 analyses related to PSV and PORV water relief.

Shearon Harris

On October 12, 2001, the NRC granted a license amendment to the Shearon Harris nuclear power plant for steam generator replacement and a power uprate to a maximum power level of 2900 MWt (approximately 4.5 percent). In addressing the licensee's evaluation of SRP Section 15.5.1, the NRC staff found that the analysis showed that the calculated inlet pressures and temperatures required for the PORVs and SRVs to operate in a water environment are within the valve operable ranges, and thus ensure that the PORV and SRV are operable during the transient. The valve operable ranges were previously determined by the licensee to support operability of the PORV and SRV during the discharge of subcooled water in accordance with the TMI Action Plan Item II.D.1 requirements. Based on the analysis meeting the acceptance criteria of SRP Section 15.5.1 with respect to the RCS pressure limit and departure-from-nucleate-boiling limit, the NRC staff concluded that the analysis was acceptable.

Beaver Valley

On July 19, 2006, the NRC granted an EPU to Beaver Valley Units 1 and 2 (BVPS-1 and 2) for an approximate 8-percent increase in thermal power to 2,900 MWt. In its SER (ADAMS No. ML061720376), the NRC staff stated that a specific issue which was reviewed related to the capability of the PSVs to discharge liquid and adequately reseat for a spurious SI actuation. The specific issue which the NRC staff evaluated in this regard was whether the PSVs could reasonably be expected to reseat in order to prevent the spurious SI actuation (a Condition II event) from causing a stuck-open PSV (a Condition III event). This issue was said to be further discussed in RIS 2005-29. While the PSVs are qualified to discharge steam, if the valves discharge liquid having a temperature low enough, they may not reseat properly.

Based the licensee's analysis, during a spurious SI event, the PSVs would be required to discharge steam followed by high temperature liquid after the pressurizer fills. The licensee provided plots of the pressurizer water temperatures for this event which indicated that the minimum temperature of the discharged liquid for both BVPS-1 and 2 is approximately 620 °F. To evaluate the capability of the valves to discharge and reseat, the NRC staff reviewed the available data from the full flow tests performed during the EPRI test program in 1981 for the specific PSV models representative of those installed at BVPS-1 and 2. The licensee also used the methodology contained in WCAP-11677, and determined that the minimum acceptable liquid temperature for which the PSVs are expected to successfully discharge and reseat is less than the minimum expected temperature for the spurious SI event for BVPS-1 and 2.

The NRC staff agreed that both the minimum expected liquid discharge temperature and the minimum acceptable liquid temperature had been conservatively calculated. Therefore, the NRC staff determined that, for purposes of preventing the occurrence of a more serious Condition III event, there is reasonable assurance that the PSVs would adequately discharge and reseat following a spurious SI actuation. A consideration in making this finding was that, in the unlikely event of a stuck-open PSV, the ECCS is fully capable of mitigating the resulting LOCA.

Turkey Point

On June 15, 2012, the NRC granted an EPU for Turkey Point, Units 3 and 4 that increased the thermal power level of each unit approximately 15 percent to 2644 MWt.

In the SER (ADAMS Accession No. ML11293A359), the NRC staff indicated that ECCS actuation is not a possible initiator of inadvertent increase in reactor coolant inventory because the high head SI pumps have a shut-off head below the normal RCS operating pressure. The NRC staff stated that a CVCS malfunction that increases RCS inventory was evaluated for the effects of adding water inventory to the RCS. If the pressurizer fills and causes water to be relieved through the PORVs or safety valves, then these valves could stick open and create a small break LOCA. The NRC staff stated that this would violate the acceptance criterion that prohibits the escalation of an anticipated operational occurrence (AOO) into a more serious event. Satisfaction of this acceptance criterion is demonstrated by showing that sufficient time exists for the operator to recognize the situation and end the charging flow before the pressurizer can fill. The NRC staff concluded that the licensee's analyses of IOECCS and CVCS events adequately accounted for operation of the plant at the proposed power level.

Regarding an inadvertent opening of a pressurizer relief valve, the licensee initially proposed that the consequences of this event are bounded by the small break LOCA. The NRC staff did not accept this proposed disposition. If action is not taken to secure the open valve by either closing the PORV or its block valve, the NRC staff stated that this event could escalate to a small break LOCA, which is contrary to the non-escalation criterion. When the pressurizer becomes water solid, water begins to flow through the open PORV. If the PORV is not qualified for water relief, the NRC staff stated that it is likely the PORV will not close upon demand. In this way, the NRC staff stated that the inadvertent opening of a PORV, an AOO, becomes a small break LOCA at the top of the pressurizer, a Condition III event. The NRC staff requested that the licensee address the inadvertent opening of the PORV with respect to the third criterion for a Condition II event.

The licensee provided an analysis, performed largely in accordance with NRC-approved, Westinghouse analytic methodology using the RETRAN computer code; however, this analysis was performed assuming that the PORV opened instead of the PSV. The NRC staff stated that assuming the opening of the PORV is acceptable, because the PSV is differently qualified, and reseats mechanically. An additional independent fault would be required to cause the safety valve to fail to close. The analysis indicated that the pressurizer would fill within about 240 seconds. The licensee stated that there are multiple alarms to indicate the opening of a PORV. The licensee stated that a prompt operator action is required to close the PORV and, if the PORV does not close, the operator is to close the block valve. Because the necessary actions are prompt and simple, the NRC staff agreed that there is sufficient time to secure the inadvertently open PORV without filling the pressurizer.

St. Lucie

On September 24, 2012, the NRC granted an EPU for St. Lucie, Unit 2 that increased the authorized thermal power level about 12 percent to 3020 MWt. Regarding an IOECCS event, the high pressure SI pumps are incapable during power operations of delivering flow to the RCS because the pumps' shut-off head is less than the normal RCS operating pressure of 2250 psia. Therefore, the inadvertent operation of the ECCS at power event is not a credible event and was not analyzed by the licensee for the proposed EPU. The NRC staff found that the licensee's position for not analyzing the IOECCS event to be acceptable.

Regarding a CVCS malfunction, this event increases RCS inventory as an AOO that is evaluated for the effects of adding water inventory to the RCS. The NRC staff reviewed the licensee's analyses of the CVCS malfunction event and concluded that the licensee's analyses adequately accounted for operation of the plant at the proposed power level and were performed using acceptable analytical models. The NRC staff determined that the licensee's analysis demonstrated that the pressurizer did not become water solid, assuring no water was discharged through the PSVs.

Regarding an IOPORV event, the NRC staff stated that when viewed from the mass addition perspective, this event can be evaluated in two phases: (1) an inadvertent opening of a pressurizer relief valve, followed by (2) an inadvertent ECCS actuation. In the first phase, the NRC staff stated that this event could be mitigated by closing the open pressurizer relief valve or its block valve. If the PORV or its block valve was not closed, the NRC staff stated that the IOPORV event would enter the second phase with actuation of the ECCS. Based on its review, the NRC staff determined that the pressurizer overfill analysis, available alarming system, and procedures in combination with simulator exercise result had provided reasonable assurance that the pressurizer would not be expected to fill to a water solid condition that could prevent the PORV or PSV from closing after they were open, and thus, supported that the event would not generate a more serious plant conditions, meeting the ~~the~~ non-escalation criterion. The NRC staff stated that it reviewed the licensee's analyses of the inadvertent opening of a pressurizer pressure relief valve event, and concluded that the licensee's analyses adequately accounted for operation of the plant at the proposed power level and were performed using acceptable analytical models.

The NRC staff concluded that the licensee demonstrated that the all AOO acceptance criteria are satisfactorily met.

North Anna

In UFSAR (Revision 51, dated September 30, 2015) Section 15.2.14, "Spurious Operation of the Safety Injection System at Power," the licensee for North Anna Units 1 and 2 discusses the plant response to an inadvertent SI event. In particular, UFSAR Section 15.2.14.2.3, "Event Propagation," states the following:

> Safety valve (Reference 18) and PORV (Reference 19) testing has revealed no instances of failure of the valves to reseat following water relief. Resulting leakage is within the capacity of the normal makeup system and is therefore not considered to be a small break loss of reactor coolant event. Therefore, the complete filling of the pressurizer and/or water relief via a safety valve as a result of a spurious safety injection does not constitute a failure to meet the event propagation acceptance criterion. Although primary credit for preventing the propagation of the event to a small break loss of reactor coolant event is the reseating of the PORVs and safety valves, it is noted that the PORVs (which open prior to the safety valves and, if open, preclude safety valve actuation for this event) are provided with block valves which the operator will close in the event of excessive PORV leakage.

North Anna UFSAR Section 15.2.14.3, "Conclusions," states that the complete filling of the pressurizer and/or water relief via a safety valve as a result of a spurious safety injection does not constitute a failure to meet the event propagation acceptance criterion. In UFSAR Section 15.2, "References," lists Reference 18 as EPRI NP-2770-LD, Volumes 3 and 4, "EPRI/CE PWR Safety Valve Test Reports for Dresser Safety Valve Models 31739A and 31709NA," February and March 1983; and Reference 19 as EPRI NP-2670-LD, Volume 6, "EPRI/Wyle Power-Operated Relief Valve Phase III Test Report, October 1982.

Conclusion

In conclusion, the reliance by the Braidwood/Byron licensee on the acceptable performance of the PSVs and PORVs for liquid service in response to abnormal events is not inconsistent with similar approaches by some other nuclear power plant licensees. In general, the review of activities by various nuclear power plant licensees related to PSV and PORV performance revealed reliance on EPRI, Wyle, and valve vendor testing to provide support for the performance of these valves under various service conditions. Specific certification for flow capacity of these valves for liquid service in accordance with the ASME BPV Code and National Board was not identified in the review of various justifications prepared by nuclear power plant licensees.

However, the Braidwood/Byron licensee has not addressed several potential safety and operational issues in support of its reliance on the performance of the PSVs and PORVs for the service conditions specified in the UFSAR. These issues include the following:

Comment [CT]: How should these issues be discussed in the main report?

1. In NSAL-93-013, Westinghouse raised a potential safety concern regarding water relief through pressurizer valves. In an LAR dated May 29, 1998, proposing to upgrade the PORVs at Braidwood and Byron, the licensee stated that "the PSRVs have not been qualified to reseat after passing subcooled liquid." The licensee later withdrew this proposed LAR. However, the actions by the Braidwood/Byron licensee to address the potential safety concern raised in NSAL-93-013 to avoid water relief through PSVs (such as performed by licensees of other nuclear power plants) are not apparent.

2. The Braidwood/Byron UFSAR states that the performance of the pressurizer safety relief valve system and the loads on pressurizer safety relief valves, associated piping, and supports as a result of liquid discharge through the pressurizer safety relief valves, was determined to be acceptable. In support of this statement, the Braidwood/Byron UFSAR references NRC SERs dated 1988 that focused on EPRI valve testing conducted in the early 1980s in response to NUREG-0737, Item II.D.1. The licensee should discuss its current justification for determining that the pressurizer valves are capable of performing their functions consistent with the assumptions for their operating conditions described in the UFSAR. For example, the licensee should indicate the positions of the reactor system designer and applicable valve manufacturers for the performance of the pressurizer valves assumed in the UFSAR. The licensee should describe its evaluation of more recent EPRI studies that discuss the potential for failure of PSVs during liquid service based on unstable test results during the EPRI testing in the 1980s. See EPRI TR-1011047 (August 2004), "Probability of Safety Valve Failure-to-Reseat Following Steam and Liquid Relief - Quantitative Expert Elicitation," that states in Appendix B that "[b]ecause these valves are not designed for liquid flow, and because EPRI tests with subcooled liquid led to unstable conditions more often than not, the likelihood of PSV failure during an SBO [station blackout] accident would be quite high."

3. The Braidwood and Byron IST Programs specify periodic fail safe tests, exercising, and position verification testing for the PORVs; and periodic position verification testing and relief valve testing in accordance with the ASME *Code for Operation and Maintenance of Nuclear Power Plants* (OM Code), Appendix I, "Inservice Testing of Pressure Relief Devices in Light-Water Reactor Nuclear Power Plants," for the PSVs. The Braidwood and Byron IST Programs should address the IST provisions for the PSVs and PORVs consistent with the assumptions for their service conditions described in the UFSAR.

4. The Braidwood and Byron IST Programs specify exercising and position verification for the PORV block valves. In addition, the Byron IST Program specifies testing using ASME OM Code Case OMN-1, "Alternative Rules for Preservice and Inservice Testing of Active Electric Motor Operated Valve Assemblies in Light-Water Reactor Power Plants," for the PORV block valves. The licensee should verify that the PORV block valves are capable of closing to isolate the PORVs consistent with the assumptions for their service conditions described in the UFSAR.

APPENDIX D: REFERENCES

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2. **ANS 2016:** ANS, "American Nuclear Society Standards Committee Glossary of Definitions and Terminology," dated May 19, 2016. Available from ANS at http://cdn.ans.org/standards/resources/toolkit/docs/glossary-of-definitions.pdf.

3. **ComEd 1982:** Commonwealth Edison (ComEd), letter from T. R. Tramm to Harold R. Denton, U.S. Nuclear Regulatory Commission (NRC), "Byron Station Units 1 and 2; Braidwood Station Units 1 and 2; Pressurizer Safety and Relief Valves," dated October 26, 1982. [This document is not in ADAMS, but is available through the NRC Public Document Room using Accession No. 8211020633 and microfiche location 15886:171 – 15886:231.]

4. **ComEd 1998:** Commonwealth Edison Company, letter from K.L. Grasser to U.S. NRC, "Change to Credit Automatic PORV Operation for Mitigation of Inadvertent Safety Injection at Power Accident," dated May 29, 1998. [This document is not in ADAMS, but is available through the NRC Public Document Room using Accession No. 9806020180 and microfiche location A3765:275-301.]

5. **ComEd 1999:** Commonwealth Edison Company, letter from R.M. Krich to U.S. NRC, "Response to Request for Additional Information Regarding License Amendment Request to Credit Automatic Power-Operated Relief Valve (PORV) Operation for Mitigation of Inadvertent Safety Injection at Power Accident and Withdrawal of License Amendment Request," dated July 16, 1999. [This document is not in ADAMS, but is available through the NRC Public Document Room using Accession No. 9907200183 and microfiche location A8671:349-354.]

6. **ComEd 2000a:** Commonwealth Edison Company, letter from R.M. Krich to U.S. NRC, "Request for a License Amendment to Permit Uprated Power Operations at Byron and Braidwood Stations," dated July 5, 2000. ADAMS Accession No. ML003730608.

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8. **Consumers 1982:** Consumers Energy Company, letter from D. P. Hoffman to Harold R. Denton, U.S. NRC, "Safety and Relief Valve Test Report for the EPRI PWR Safety and Relief Valve Test Program," dated April 1, 1982. [This document is not in ADAMS, but is available through the NRC Public Document Room using Accession No. 8207160337 and microfiche location 13866:001 – 13869:106.]

9. **EPRI 1982a:** Electric Power Research Institute (EPRI), NP-2770-LD, "EPRI/C-E PWR Safety Valve Test Report," dated January 1982. [Publicly available from EPRI for a fee.]

10. **EPRI 1982b:** EPRI, NP-2670-LD, "EPRI/Wyle Power-Operated Relief Valve Phase III Test Report," dated October 1982. [Publicly available from EPRI for a fee.]

11. **EPRI 1982c:** EPRI, NP-2628-SR, "EPRI PWR Safety and Relief Valve Test Program – Safety and Relief Valve Test Report," dated December 1982. [This document is not in ADAMS, but is available through the NRC Public Document Room using Accession No. 8407130197 and microfiche location 25588:082-262.]

12. **EPRI 2004:** EPRI, TR-1011047, "Probability of Safety Valve Failure-to-Reseat Following Steam and Liquid Relief - Quantitative Expert Elicitation," dated August 2004. Publicly available from EPRI.

13. **Exelon 2001:** Exelon Generation Company, LLC, letter from R.M. Krich 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," dated January 31, 2001. ADAMS Accession No. ML010330145.

14. **Exelon 2002:** Exelon Generation Company, LLC, letter from R. M. Krich to U.S. NRC, "Byron/Braidwood Stations Updated Final Safety Analysis Report, Revision 9; Byron Station Technical Requirements Manual, Revision 32; Byron Station Technical Specifications Bases, Revision 31; Braidwood Station Technical Requirements Manual, Revision 26; Braidwood Station Technical Specifications Bases, Revision 36," dated December 16, 2002. [Non-publicly available; electronic files are in the NRC file center, and the transmittal letter is at ADAMS Accession No. ML023650683.]

15. **Exelon 2014:** Exelon Generation Company, LLC, "Byron/Braidwood Nuclear Stations Updated Final Safety Analysis Report (UFSAR)," Revision 15, dated December 2014. ADAMS Accession No. ML14363A393.

16. **Exelon 2015:** Exelon Generation Company, LLC, letter from J. Bradley Fewell to William M. Dean, U.S. NRC, "Appeal of Imposition of Backfit Regarding Compliance with Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.34(b), General Design Criteria (GDC) 15, GDC 21, GDC 29, and Licensing Basis," dated December 8, 2015. ADAMS Accession No. ML15342A112.

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25. **NRC 1981a:** U.S. NRC, NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition" [SRP], Section 15.0, "Introduction," Revision 2, dated July 1981. ADAMS Accession No. ML052350113.

26. **NRC 1981b:** U.S. NRC, NUREG-0800, Section 15.5.1 – 15.5.2, "Inadvertent Operation of ECCS and Chemical and Volume Control System Malfunction that Increases Reactor Coolant Inventory," Revision 1, dated July 1981. ADAMS Accession No. ML052350142.

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75. **PG&E 2003:** Pacific Gas and Electric Company, letter from David H. Oatley to U.S. NRC, "Response to NRC Request for Additional Information Regarding License Amendment Request 01-018, 'Credit for Automatic Actuation of Pressurizer Power Operated Relief Valves; Pressurizer Safety Valve Loop Seal Temperature,'" dated November 21, 2003. ADAMS Accession No. ML033360735

76. **Union Electric 2000:** Union Electric Company, letter from Alan C. Passwater to U.S. NRC, "Revision to Technical Specifications 3.3.2, 3.4.10, and 3.4.11 – Pressurizer Safety Valves and PORVs," dated September 25, 2000. ADAMS Accession No. ML003719636.

77. **VEPCO 2009:** Virginia Electric and Power Company (VEPCO), letter from L.N. Hartz to U.S. NRC, "North Anna Power Station Units 1 and 2 Updated Final Safety Analysis Report Revision 45," dated October 1, 2009. ADAMS Accession No. ML092810047.

78. **VEPCO 2015:** Virginia Electric and Power Company, letter from Gianna C. Clark to U.S. NRC, "North Anna Power Station Units 1 and 2 Updated Final Safety Analysis Report Revision 51," dated September 30, 2015. ADAMS Accession No. ML15296A098 [non-public].

79. **Westinghouse 1988:** Westinghouse, R.J. Dickinson and J.G. Bass, "Pressurizer Safety Relief Valve Operation for Water Discharge during a Feedwater Line Break," WCAP-11677, dated January 1988. [This document is not in ADAMS, but is available through the NRC Public Document Room using Accession No. 8905120191 and microfiche location 49755:336 – 49756:017.]

80. **Westinghouse 1993:** Westinghouse, Nuclear Safety Advisory Letter (NSAL) 93-013, "Inadvertent ECCS Actuation at Power," dated June 30, 1993. [This document is not in public ADAMS, but is available through the NRC Public Document Room using Accession No. 9608200112 and microfiche location 89419:311-315, as well as in non-public ADAMS Accession No. ML052930330, pages 2-6 of file.]

81. **Westinghouse 1994:** Westinghouse, NSAL-93-013, Supplement 1, "Inadvertent ECCS Actuation at Power," dated October 28, 1994. ADAMS Accession No. ML050320117 [pages 9-15 of file].

82. **Westinghouse 2000:** Westinghouse, NSAL-00-013, "CVCS Modeling Assumption for Loss of Offsite Power Analyses," dated August 23, 2000. ADAMS Accession No. ML103200150 [pages 131-139 of file].

83. **Westinghouse 2007:** Westinghouse, NSAL-07-10, "Loss-of-Normal Feedwater Loss-of-Offsite AC Power Analysis PORV Modeling Assumptions," dated November 7, 2007. ADAMS Accession No. ML100140163 [pages 23-26 of file]

84. **WOG 1982:** Westinghouse Owners Group (WOG), letter from Oliver D. Kingsley, Alabama Power Company, to Harold R. Denton, U.S. NRC, "NUREG-0737, Item II.D.1, 'Pressurizer Safety Valve Operability,'" dated July 27, 1982. Forwards Westinghouse WCAP-10105, "Review of Pressurizer Safety Valve Performance as Observed in the EPRI Safety and Relief Valve Test Program," dated June 1982. [This document is not in ADAMS, but is available through the NRC Public Document Room using Accession No. 8208190307 and microfiche location 14387:189-301.]

From:	Clark, Theresa
To:	Holahan, Gary; West, Steven; Scarbrough, Thomas; Spencer, Michael
Subject:	REVIEW: informal concurrence version of panel report
Date:	Sunday, August 21, 2016 12:06:29 AM
Attachments:	Backfit Appeal Panel Report (MASTER).docx Backfit Appeal Panel Report (MASTER) - tracked.docx cover memo (MASTER).docx
Importance:	High

Hi all—attached are the cover memo (no change since the last version, I think) and report (both clean and with changes tracked to the last version I sent—not since the beginning). I incorporated the edits that Michael sent Friday.

I recommend that you guys read these and send any remaining edits/comments by mid-day Tuesday—consider this informal concurrence. Then I can give another look before we have the meeting with Vic. Shortly thereafter I assume we would be able to sign an official copy. (Somewhere in there I will ask Patti to make a concurrence package.)

Thanks!

--

Theresa Valentine Clark
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U.S. Nuclear Regulatory Commission
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Report of the Backfit Appeal Review Panel
Chartered by the
Executive Director for Operations to
Evaluate the June 2016 Exelon Backfit Appeal

August XX, 2016

ADAMS Accession No. MLXXXXXXXXX

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1 BACKGROUND

On June 22, 2016,[1] in accordance with NRC Management Directive (MD) 8.4,[2] the NRC Executive Director for Operations (EDO) established a Backfit Appeal Review Panel (Panel) to review the appeal by Exelon Generation Company, LLC (Exelon or the licensee) of the U.S. Nuclear Regulatory Commission (NRC) staff's determination that a backfit is necessary at Braidwood Station, Units 1 and 2 (Braidwood) and Byron Station, Units 1 and 2 (Byron), as well as the NRC staff's application of the compliance backfit exception provided in Title 10 of the *Code of Federal Regulations* (10 CFR), Section 50.109, "Backfitting."

This backfit determination is documented in an October 9, 2015, letter (referred to as the Backfit Letter).[3] The letter describes the NRC staff's review of licensing basis documents for Byron and Braidwood. The NRC staff determined that Byron and Braidwood were not in compliance with the plant-specific design bases and several NRC regulations:

- General Design Criterion (GDC) 15, "Reactor coolant system design," in 10 CFR Appendix A, "General Design Criteria for Nuclear Power Plants"

- GDC 21, "Protection system reliability and testability"

- GDC 29, "Protection against anticipated operational occurrences"

- Paragraph (b) of 10 CFR 50.34, "Contents of applications; technical information"

Specifically, the NRC staff determined that Byron and Braidwood do not comply with provisions in American Nuclear Society (ANS) Standard 51.1/N18.2-1973[4] for ensuring that ANS Condition II events[5] do not progress to more serious ANS Condition III events following water discharge[6] through certain valves. The NRC staff acknowledged that the NRC staff position differed from a previous staff position documented in a May 4, 2001, safety evaluation (SE) supporting a stretch power uprate (referred to as the Uprate SE).[7] However, the NRC staff determined that the backfitting was justified under the compliance exception in 10 CFR 50.109(a)(4)(i). The licensee was directed to take action to resolve the non-compliance.

On December 8, 2015, the licensee appealed the NRC staff's decision to the Director of the Office of Nuclear Reactor Regulation (NRR), stating its disagreement with the NRC's conclusion that the compliance exception to the backfit rule applies in this case, and that the NRC has twice approved the underlying analysis.[8] The referenced approvals were an August 26, 2004, license amendment associated with pressurizer safety valve (PSV) setpoints[9] and the above-

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[1] NRC 2016e (Author and year citations in footnotes refer to the designation of references in Appendix D to this report.)
[2] NRC 2013
[3] NRC 2015b – referred to as the Backfit Letter in the remainder of the report
[4] ANS 1973
[5] Specifically, inadvertent operation of the emergency core cooling system (IOECCS), malfunction of the chemical and volume control system (CVCS), and inadvertent opening of a pressurizer safety or relief valve.
[6] For consistency in this report, the Panel uses the phrase "water discharge" rather than "water relief" or "liquid discharge" (except in direct quotes), as this is the phrase used in the Westinghouse documents that raised the issue addressed in this report.
[7] NRC 2001b – referred to as the Uprate SE in the remainder of the report
[8] Exelon 2015 – referred to as the NRR Appeal in the remainder of the report

referenced Uprate SE. In a letter dated May 3, 2016, the NRC responded to the licensee's appeal and reaffirmed its decision that the backfit per the compliance exception provisions of 10 CFR 50.109(a)(4)(i) is appropriate.[10]

On June 2, 2016, the licensee again appealed the NRC staff's decision, this time to the EDO.[11] The purpose of this report by the Backfit Appeal Review Panel is to provide information and recommendations to support the decision of the EDO.

1.1 Conduct of the Panel's Review

In order to establish a technically sound, well informed, and legally defensible basis for its recommendations, the Backfit Appeal Review Panel undertook a review of the relevant documents in this case. This included the licensee and NRC staff letters mentioned above; the Uprate SE and Setpoint SE; and a June 16, 2016, letter from the Nuclear Energy Institute (NEI)[12] supporting the Exelon backfit appeal. The Panel also reviewed many other related documents, which fall into five broad categories:

- The Backfit Rule (10 CFR 50.109), related court actions, and Commission and staff guidance on application of the Backfit Rule

- Docketed communications for Byron and Braidwood from 1982 to the present, including license amendment requests (LARs) by the licensee, NRC-issued license amendments, NRC requests for additional information (RAIs), licensee responses, meeting summaries, NRC SEs, and the licensee's Updated Final Safety Analysis Report (UFSAR)

- NRC guidance relevant to the analysis of IOECCS events over the period of 1981 to the present, including Standard Review Plan (SRP) Section 15.0, Section 15.5.1 – 15.5.2, and Section 15.6.1[13]

- Westinghouse Nuclear Safety Advisory Letter (NSAL) 93-013[14] and its Supplement 1[15], as well as docketed communications regarding actions taken by other licensees in response to Westinghouse NSAL-93-013

- The history of NRC and industry activities related to power operated relief valves (PORVs), their block valves, and PSVs (including Three Mile Island (TMI) Action Plan Items II.D.1, II.D.3,II.G.1, II.K.3 documented in NUREG-0737[16], as well as Generic Letter 89-10[17] and its supplements), Electric Power Research Institute (EPRI) valve testing, and operating experience (NUREG/CR-7037[18])

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[9] NRC 2004b – referred to as the Setpoint SE in the remainder of the report
[10] NRC 2016d – referred to as NRR Appeal Decision in the remainder of the report
[11] Exelon 2016a – referred to as EDO Appeal in the remainder of the report
[12] NEI 2016
[13] NRC 1981a, NRC 1981b, NRC 1981c, NRC 2007a, NRC 2007b, and NRC 2007c
[14] Westinghouse 1993
[15] Westinghouse 1994
[16] NRC 1980c – referred to as the TMI Action Plan in the remainder of the report; lessons learned from TMI were also presented in NUREG-0578 (NRC 1979a), NUREG-0585 (NRC 1979b), and NUREG-0660 (NRC 1980a)
[17] NRC 1989
[18] NRC 2011

In addition to the document review, the Panel had the benefit of meetings with NRR (both the Division of Safety Systems and the Division of Engineering), the Office of the General Counsel, and the NRC Committee to Review Generic Requirements (CRGR). Both Exelon (Bradley Fewell, Senior Vice President of Regulatory Affairs) and NEI (Tony Pietrangelo, Senior Vice President and Chief Nuclear Officer) declined offers for a public meeting, but indicated a willingness to provide information if the Panel identified the need. The Panel did not identify a need for additional information from either Exelon or NEI to complete its review, which is summarized below and documented in the attached report.

At the request of the Panel, the Office of Nuclear Regulatory Research (RES) conducted risk analyses using the NRC's Standardized Plant Analysis Risk model for Byron Unit 1.[19] These analyses informed the Panel's response to the question from the EDO regarding the risk significance of the relevant accident sequences.

1.2 Proposed Compliance Backfit and Exelon Appeals

In the Backfit Letter, the NRC staff informed Exelon that it had determined that Byron and Braidwood are not in compliance with GDCs 15, 21, and 29; 10 CFR 50.34(b); and the plant-specific design bases that were expected to demonstrate there will be no progression of Category II events to Category III events. The NRC staff stated that based on its review of Byron and Braidwood UFSAR Sections 15.5.1, 15.5.2, and 15.6.1, the UFSAR predicts water discharge through a valve that is not "qualified" for water discharge. Therefore, the NRC staff concluded that the UFSAR does not contain analyses that demonstrate that the plants' structures, systems, and components (SSCs) will meet the design criteria for ANS Condition II faults as stated in Byron and Braidwood UFSAR Section 15.0.1.2. Based on the SE attached to its letter,[20] the NRC staff found that the licensee must take action to resolve the non-compliance.

The Backfit SE addressed three accident analyses in Chapter 15 of the Byron and Braidwood UFSAR: (1) IOECCS; (2) CVCS malfunction that increases reactor coolant inventory; and (3) inadvertent opening of a pressurizer safety or relief valve (IOPORV). The NRC staff noted that each ANS Condition II event must be shown to meet the following:

1. no fuel damage,
2. no overpressure of the reactor coolant system (RCS) or main steam system, and
3. no progression into an event of a more serious category without another independent fault.

Regarding an IOECCS, the NRC staff stated in Section 3.1.2.1 of the Backfit SE that use of the block valve to isolate a stuck-open PORV was unacceptable. The NRC staff stated that Westinghouse recommended this approach in 1993 and that the NRC staff rejected this approach in 2005 (RIS 2005-29[21]).

In Section 3.1.2.4 of the Backfit SE, the NRC staff stated that the Byron and Braidwood IOECCS analysis depends on water discharge through the PSVs. The NRC staff faulted the

[19] NRC 2016f
[20] Referred to as the Backfit SE in the remainder of the report.
[21] NRC 2005b

licensee for “not appl[ying] the single-failure assumption” and stated that the following information was necessary to support water qualification of the PSVs:

1. In accordance with the American Society of Mechanical Engineers (ASME) *Boiler and Pressure Vessel Code* (BPV Code), Section III, provide the original Overpressure Protection Report defining operating conditions and required relief capacities, and manufacturer’s certification and test results

2. In accordance with the ASME *Code for Operation and Maintenance of Nuclear Power Plants* (OM Code), provide inservice test history for PSVs, including water and steam tests, or provide correlation test for alternative test fluid.

Regarding a CVCS malfunction, the NRC staff stated in Section 3.2 of the Backfit SE that the licensee had not provided an analysis for the CVCS malfunction that increases reactor coolant inventory that demonstrates the plants’ ability to meet the requirements of an ANS Condition II event.

Regarding an IOPORV, the NRC staff stated in Section 3.3 of the Backfit SE that the licensee had not provided an analysis for the IOPORV that extends long enough into the transient to demonstrate the event would not transition from an ANS Condition II event to an ANS Condition III event.

In the Backfit SE, the NRC staff referenced Millstone[22] and Callaway[23] license amendments as examples of licensees upgrading PORVs for water discharge; a Beaver Valley extended power uprate (EPU) license amendment[24] as an example of qualifying PORVs for water discharge; and Turkey Point[25] and St. Lucie Unit 2[26] EPU amendments as additional precedent in support of the backfit decision.

In the NRR Appeal, Exelon asserted that the NRC had not justified invoking the compliance exception to the backfit rule. Exelon stated that the NRC approved its IOECCS analysis in the Uprate SE and Setpoint SE.

In the NRR Appeal Decision, the NRC staff stated that the previous approvals were inconsistent with the Agency’s general position on the known and established standard at issue, in this case the progression of ANS Condition II events to higher level events. The NRC staff stated that the fact that the NRC staff were aware of references to EPRI reports on the ability of these non-water qualified PSVs to reseat in certain circumstances is not sufficient to support the licensee’s position.

In the EDO Appeal, Exelon stated that the NRC had misidentified the “known and established standard” at issue as the prohibition of ANS Condition II events progressing to ANS Condition III events. Exelon asserted that the standard in question concerns what is necessary to “qualify” valves for water discharge. Exelon contended that this standard is the EPRI testing and analysis, and that the NRC has agreed that Byron and Braidwood meet this standard. Exelon also contended that the change in NRC staff position on prior approvals is not a mistake of fact,

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[22] NRC 1998
[23] NRC 2000
[24] NRC 2006
[25] NRC 2012a
[26] NRC 2012b

but rather a new or modified interpretation of compliance with NRC requirements, for which use of the compliance exception provided for in the Backfit Rule is not appropriate.

1.3 Backfit Rule and the Compliance Exception

Backfitting is defined by 10 CFR 50.109(a) as:

> … the modification of or addition to systems, structures, components, or design of a facility; or the design approval or manufacturing license for a facility; or the procedures or organization required to design, construct or operate a facility; any of which may result from a new or amended provision in the Commission's regulations or the imposition of a regulatory staff position interpreting the Commission's regulations that is either new or different from a previously applicable staff position … .

Unless one of three specified exceptions apply, the NRC may impose a backfit only if it performs a backfit analysis in accordance with 10 CFR 50.109(a)(2) and determines in accordance with 10 CFR 50.109(a)(3) "that there is a substantial increase in the overall protection of the public health and safety or the common defense and security to be derived from the backfit and that the direct and indirect costs of implementation for that facility are justified in view of this increased protection."

Section 50.109(a)(4) sets forth the three exceptions to the requirements of 10 CFR 50.109(a)(2) and (a)(3). The first exception, the compliance exception, applies if the "modification is necessary to bring a facility into compliance with a license or the rules or orders of the Commission, or into conformance with written commitments by the licensee." The second and third exceptions relate to actions necessary to ensure adequate protection or to actions that involve defining or redefining adequate protection.

The Commission explained its intended application of the compliance exception in the Statements of Consideration (SOC) accompanying the 1985 final rule amending 10 CFR 50.109:[27]

> The compliance exception is intended to address situations in which the licensee has failed to meet known and established standards of the Commission because of omission or mistake of fact. It should be noted that new or modified interpretations of what constitutes compliance would not fall within the exception and would require a backfit analysis and application of the standard.

In the same SOC, the Commission acknowledged that staff interpretations of rules are not legally binding, but the Commission also stated that "staff interpretations of broadly stated rules are often necessary to give a rule effect and in some instances may be a causal factor in initiating a backfit."[28]

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[27] NRC 1985, at 38103

[28] NRC 1985, at 38102. The 1985 backfit rule was vacated by a Federal court on grounds unrelated to the compliance backfit exception. See *Union of Concerned Scientists v. U.S. Nuclear Regulatory Com'n*, 824 F.2d 108, 119-20 (1987). In 1988, the Commission amended the backfit rule (NRC 1988b) to address the court's concerns, but did not change the 1985 rule's compliance exception provision. Thus, the quoted statements from the 1985 rule are the applicable expression of Commission intent regarding compliance backfits.

By its terms, the compliance exception applies to actions necessary for compliance with rules, licenses, and orders, or for conformance with written commitments.[29] Also, the Commission explicitly acknowledged the importance of staff interpretations of rules in the regulatory process. Thus, the Panel understands the term "known and established standard" to include standards established in rules, licenses, orders, and written commitments, and NRC interpretations of rules. Some standards may be broad-based, while others may apply only to a limited number of plants. As stated in NUREG-1409, "[i]nformal or formal communications to one licensee are not official positions to all licensees. ... Orders, licenses, and written commitments are applicable only to a particular licensee."

The failure to meet a known and established standard is grounds for a compliance backfit if this failure is due to "omission or mistake of fact." Thus, if a licensee obtains NRC approval of an alternative to a specific standard set forth in guidance, that standard and guidance could not be used to support a compliance backfit unless the NRC's approval of the alternative was based on an omission or mistake of fact. "Known and established standards" are to be distinguished from "new or modified interpretations of what constitutes compliance," which do not fall within the compliance exception. The Panel understands the term "new or modified interpretations" to include situations where the NRC has, in effect, "changed its mind" on how to interpret the language of a requirement or on how much assurance is necessary to conclude that the requirement is met. Levels of assurance might be established in terms such as acceptable probabilities or consequences, conservative assumptions, or sufficient margin.

Additional background information on the Backfit Rule and the compliance exception is provided in Appendix A to this report.

1.4 A Brief History of Pressurizer Valve Issues

Appendix B to this report provides a summary of the NRC and industry's testing, evaluation, and other consideration of PORVs and PSVs since the TMI Unit 2 (TMI-2) accident in 1979. This historical review provides context for discussion of valve "qualification" in the Backfit SE. It also provides the basis for the Panel's conclusions regarding the "known and established standard" for "qualification" in the context of the TMI Action Plan item and subsequent activities, as well as how it should be interpreted in the Byron and Braidwood licensing basis.

In light of the NRC staff's assertion that the licensee had not applied the "single-failure assumption" as noted above, the Panel considered the applicability of the single failure criterion to PSVs. The Panel expended considerable effort in searching for an answer to what appears to be a simple question: "Are PSVs active components subject to the single failure criterion, or are they passive components exempt from it?" NRR staff have taken the position that PSVs have consistently been treated as active components.

In the Panel's evaluation of the treatment of PSV failure potential (Section 3 below), an historical perspective is provided. In general, the Panel found that the classification of a component as "active" or "passive" depends on its design, application, and function. For example, passive components almost always do not need external power; usually do not need an external actuator (e.g., signal)[30]; sometimes do not involve any mechanical motion (e.g., movement of a

[29] NUREG-1409 (NRC 1990c) defines written commitments broadly to include the "final safety analysis report, licensee event reports, and docketed correspondence, including responses to NRC bulletins, generic letters, inspection reports, or notices of violation and confirmatory action letters."
[30] For example, SECY-77-439 (NRC 1977) states: "Examples [of passive failures in fluid systems] include

valve disc)[31]; and sometimes do not involve any motion, either fluid or mechanical (e.g., piping). International Atomic Energy Agency (IAEA) TECDOC-1624 states that "[s]afety related terms such as passive and inherent safety have been widely used, particularly with respect to advanced nuclear plants, generally without definition and sometimes with definitions inconsistent with each other." This guidance further defines four level of "passivity" to "to help eliminate confusion and misuse of the terms by members of the nuclear community." In addition, SECY-05-0138[32] also acknowledges and discusses inconsistencies in the use and application of the term "passive."

The introduction to the GDCs and the related footnote define the applicability of the single failure criterion in terms of electrical versus fluid systems, and active versus passive components. Neither the GDCs nor NRC guidance define which characteristics of passive components are necessary to make a component exempt from the single failure criterion. Some examples are clear: pipes are passive components and pumps and motor-operated valves that operate to perform their safety functions are active components. As discussed in Section 3.6 below, check valves might be classified as active or passive components depending on specific considerations.

With respect to PSVs, the ASME BPV Code applicable to Byron and Braidwood includes requirements for overpressure protection that relate to the single failure criterion through several specific design and construction requirements. As a result, the PSVs are conservatively sized with sufficient margin to accommodate a single failure although the single failure criterion is almost never explicitly discussed or applied in accident analyses. The Byron and Braidwood UFSAR states that "adequate overpressurization protection is provided by the three installed safety valves." Neither the UFSAR system descriptions nor the safety analyses provide detailed discussions of potential PSV failures or their consequences. The principal discussion of potential PSV failures in the accident analyses occurs in the evaluation of an inadvertent opening of a PSV in UFSAR Section 15.6.1.

Most relevant for the current issue, the Byron and Braidwood UFSAR analyses of overpressure events (e.g., loss of load, loss of feedwater) do not apply the single failure criterion to cause a PSV to stick open (i.e., fail to reseat) when opening on steam flow. In addition, the UFSAR Feedwater System Pipe Break analysis (Chapter 15.2.8) does not apply the single failure criterion to cause a PSV to stick open either during steam discharge or during water discharge. A survey of other Westinghouse-designed plants showed that this treatment of PSV valve performance during anticipated operational occurrences (AOOs, similar to ANS Condition II events) and postulated accidents (similar to ANS Condition IV events) has been consistent and without any identified exceptions.[33]

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the failure of a simple check valve to move to its correct position when required, the leakage of fluid from failed components, such as pipes and valves—particularly through a failed seal at a valve or pump—or line blockage. Motor-operated valves which have the source of power locked out are allowed to be treated as passive components."

[31] For example, NUREG-1800 (NRC 2001c) states that "'[p]assive' structures and components, for the purpose of the license renewal rule, are those that perform an intended function … without moving parts or without a change in configuration or properties … 'passive' may also be interpreted to include structures and components that do not display 'a change of state.'"

[32] NRC 2005a

[33] Examples include Watts Bar (NRC 1982 and TVA 1983), North Anna (NRC 1976), and AP1000 (Westinghouse 2011).

1.5 History of Westinghouse NSAL and Related Activities

Appendix C to this report provides the Panel's review of the issues identified by Westinghouse in NSAL-93-13 and its Supplement 1, how various licensees responded to these issues, and how the NRC was involved in reviewing and approving these actions. This review provides the basis for the Panel's conclusions related to the approach taken by Byron and Braidwood to address these issues in their licensing basis, as well as on the "known and established standard" for event escalation from ANS Condition II to ANS Condition III, referred to hereafter as the "non-escalation position."

2 SUMMARY OF PANEL FINDINGS

For the reasons provided in Section 3, the Panel concludes that in 2001 and 2004 and at present, the known and established standard of the Commission is 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. The Panel also concludes that, in preparing the Uprate SE and the Setpoint SE, the NRC staff exercised reasonable and well-informed engineering judgment when the NRC staff concluded that the PSVs were unlikely to stick open. The non-escalation position does not establish specific standards for valve qualification, so the non-escalation position, standing alone, provides no basis for rejecting the licensee's reliance on EPRI valve testing. Moreover, the Panel finds that no mistake or error occurred in the licensee's or previous staff's reliance on the EPRI testing program that included an evaluation of water discharge through pressurizer valves.[34] Therefore, the Panel also concludes that the position on valve qualification in the Backfit SE is a new or modified interpretation of what constitutes compliance.

3 DISCUSSION

The compliance exception to the Backfit Rule is intended to address failures to meet known and established Commission standards because of omission or mistake of fact. New or modified interpretations of what constitutes compliance do not fall within the exception. The Panel reviewed and evaluated the information referenced in this report to determine if, in 2001 and 2004, there was a known and established standard of the Commission relating to the potential for PSVs to fail following water discharge during IOECCS events.

In addition, the Panel considered the issue of "known and established standards of the Commission" as it relates to "event escalation." In the NRR Appeal Decision, the NRC staff stated that the Backfit SE "showed that the approvals at issue for Braidwood and Byron were inconsistent with the Agency's general position on the known and established standard at issue, in this case the progression of [ANS] Condition II events." The Panel recognizes that the non-escalation position, although not included in NRC regulations, is widely referenced in reactor licensing bases as an approach for addressing AOOs and postulated accidents as articulated in the GDCs. The non-escalation position is incorporated in Section 15.0.1.2 of the Byron and Braidwood UFSAR as "By definition, these faults (or events) do not propagate to cause a more serious fault, i.e., [ANS] Condition III or IV events."

[34] "Pressurizer valves" is used in this report to refer to either PORVs or PSVs when discussing issues common to both types of valves.

Neither Exelon nor the Panel disputes that the non-escalation position is now, and was in 2001 and 2004, a part of the licensing basis of both Byron and Braidwood. The Panel supports the NRC staff's view that non-escalation (from ANS Condition II to ANS Condition III or IV) is a known and established standard applicable to Byron and Braidwood. However, the Panel agrees with Exelon that the fundamental issue is not the non-escalation position, but the appropriate standard for PSV water discharge. In the absence of a PSV failure to reseat, the concerns articulated in the backfit related to event classification, event escalation, and compliance with 10 CFR 50.34(b) and GDCs 15, 21, and 29 would no longer be at issue.

The Panel's evaluation of the treatment of PSV failure potential includes an assessment of multiple relevant references, which are discussed chronologically in the sections that follow.

3.1 General Design Criteria (1971)

In 1971, the Atomic Energy Commission published the GDCs, which had been under development since 1965.[35] The introduction to Appendix A addresses "Single Failure" in the section on Definitions and Explanations. The paragraph on single failures includes a footnote stating: "The conditions under which a single failure of a passive component in a fluid system should be considered in designing the system against a single failure are <u>under development</u>" (emphasis added).

3.2 Commission Paper on Single Failure (1977)

In response to several staff concerns and differing views on the subject of application of the single failure criterion, the Acting Director of NRR issued SECY-77-439 "[t]o inform the Commission of the present status and future use of the Single Failure Criterion as a tool in the reactor safety process."[36] In part, that paper addressed the application of the single failure criterion to passive components in fluid systems, stating that "[a]pplication of the [single failure] concept is complicated by the interrelationships between the various fluid and electrical systems and their supporting auxiliaries in a nuclear power plant. Furthermore, there is a need to stipulate the events and associated assumptions which must be considered during application of the Single Failure Criterion."

SECY-77-439 specifically spoke to how "additional passive failures"—that is, failures in addition to the initiating event—had been and should be addressed, stating (with emphases added):

> During subsequent years [since the single failure footnote quoted above was published] staff <u>assumptions regarding the nature of passive failures which should be considered have not been completely consistent</u> and there has been some disagreement. However, on the basis of the licensing review experience accumulated in the period since 1969, it has been judged <u>in most instances that the probability of most types of passive failures in fluid systems is sufficiently small that they need not be assumed</u> in addition to the initiating failure in application of the Single Failure Criterion to assure safety of a nuclear power plant.

SECY-77-439 provides definitions and examples for distinguishing between active and passive failures. Among these examples, SECY-77-439 cites "the failure of a simple check valve to

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[35] AEC 1971
[36] NRC 1977

move to its correct position when required" as a passive failure. Of the examples cited in SECY-77-439, the check valve example is most similar from a mechanical perspective to the PSV failure addressed in the Backfit SE, as explained below in the discussion of SECY-94-084.

SECY-77-439 also stresses the use of engineering judgment relating to the probability of component failure and does not suggest that valve "certification" or "qualification" in accordance with ASME standards should be invoked as the basis for such decisions.

3.3 TMI Action Plan Item II.D.1 (1980)

As an element of the TMI Action Plan, the NRC staff required licensees to address the capability of relief and safety valves to perform their intended functions without failure. Specifically, Item II.D.1 states that "[p]ressurized-water reactor [PWR] and boiling-water reactor [BWR] licensees and applicants shall conduct testing to qualify the [RCS] relief and safety valves under expected operating conditions for design-basis transients and accidents." With reference to planned EPRI testing and other generic industry test programs, NUREG-0737 specified provisions for then-operating nuclear power plants and applicants for operating licenses and holders of construction permits to address the TMI Action Plan items, including Item II.D.1. NUREG-0737 stated, for the performance testing of relief and safety valves for Item II.D.1, that "[t]he testing should demonstrate that the valves will open and reclose under the expected flow conditions."

Although limited in scope, the EPRI test results did not identify any generic issues with PSVs or PORVs sticking open following water discharge. The NRC staff approvals summarized below show that the word "qualify" in this TMI Action Plan item was not intended to refer to ASME valve certification or qualification. Instead, "qualify" was used in a less formal sense to refer to a reasonable judgment that the valve would open to relieve pressure and then reliably reseat. As referenced in NUREG-0737, the EPRI test program was the widely used approach to address TMI Action Plan Item II.D.1 at PWR nuclear power plants. The Westinghouse Owners Group submitted WCAP-10105 to the NRC in 1982 to demonstrate the acceptability of the EPRI testing program for PSVs and PORVs in Westinghouse-designed PWRs.[37]

3.4 NRC Closure of TMI Action Plan Item II.D.1 for Byron and Braidwood (1988-1990)

A 1988 letter from the NRC staff to the licensee for Byron found the licensee's reliance on EPRI testing of PSVs to be acceptable.[38] The 1988 SE states that the test program was designed "[t]o reconfirm the integrity of the overpressure protection system and thereby assure that the [GDCs] are met." As discussed in Appendix B to this report, the 1988 SE describes the evaluation of the PSVs and PORVs for feedwater line break accidents that would include water discharge, and determined that the EPRI tests were applicable to the Byron and Braidwood PSVs and PORVs. Based on the NRC staff and contractor review, the 1988 SE found that the performance of the PSVs and PORVs was acceptable based on the EPRI tests.

For the specific extended high pressure injection event, the 1988 SE states that water discharge through the PSVs and PORVs could be disregarded because of the long time available for operator action. However, the SE addressed water discharge through the PSVs and PORVs as part of the feedwater line break evaluation.

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[37] WOG 1982
[38] NRC 1988c, referred to as the 1988 SE

In the cover letter for the 1988 SE, the NRC staff states that the licensee should develop and adopt plant procedures to inspect the pressurizer valves after each lift involving loop seal or water discharge. The 1988 SE contains no reference to or suggestion of a need for certification of these valves in accordance with the ASME BPV Code for water discharge capability. In 1990, the use of the EPRI test program was also found similarly acceptable for Braidwood.[39]

3.5 Westinghouse NSAL-93-013 and Supplement 1 (1993-1994)

In 1993, Westinghouse sent NSAL-93-013 to operating nuclear power plants in response to its discovery that potentially non-conservative assumptions had been used in the licensing analysis of the IOECCS event. Westinghouse recommended that licensees determine if their pressurizer safety relief valves (PSRVs)[40] "are capable of closing following discharge of subcooled water." Westinghouse noted that the PSRVs might have been designed or "qualified" to relieve subcooled water. Westinghouse also noted that "licensees may have qualified these valves in compliance to NUREG-0737, Item II.D.1." If the PSRVs were not designed or qualified for subcooled water discharge, Westinghouse recommended that licensees reevaluate the IOECCS event with three possible options of (1) reducing emergency core cooling system (ECCS) flow used in the safety analysis, (2) using a less restrictive operator response time, or (3) crediting the use of one or more PORVs to help mitigate the accident.

In Supplement 1 to NSAL-93-013, Westinghouse alerted licensees to potential reduced time for operator action if a positive displacement pump (a typical part of the CVCS) were in service, and to the need to qualify the PSRVs and the piping downstream of the PSRVs and PORVs if water discharge from the pressurizer is predicted.

Some licensees submitted license amendments that involved improvements to the PORVs and their circuitry to avoid water discharge through the PSVs (e.g., Salem[41], Millstone[42], Callaway[43], and Diablo Canyon[44]). The NRC staff review and approval of those proposed improvements relied on engineering judgment relative to the various test information and PORV circuitry upgrades described by individual licensees. The licensee for Byron and Braidwood submitted an LAR for similar PORV improvements,[45] but that request was later withdrawn.[46]

As indicated below, the Panel's sampling review found two plants, in addition to Byron and Braidwood, that chose to address this issue by crediting the capability of PSVs to relieve water, based on the EPRI testing performed in response to TMI Action Plan Item II.D.1.

3.6 Commission Paper on Passive Plant Designs (1994)

In 1994, in preparation for the design certification reviews of passive reactor designs (e.g., the Westinghouse Advanced Passive 1000 (AP1000) and the General Electric Economic Simplified

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[39] NRC 1990a
[40] Westinghouse used the term PSRVs. The specific valves for Byron and Braidwood should be designated as "safety valves" or "pressurizer safety valves" as they are by the manufacturer, in the ASME BPV Code, and by the licensee. This difference in terminology is not significant to any of the findings or conclusions in this report.
[41] NRC 1997
[42] NRC 1998
[43] NRC 2000
[44] NRC 2004a
[45] ComEd 1998
[46] ComEd 1999

Boiling-Water Reactor (ESBWR)), the NRC staff presented nine issues to the Commission for policy decisions.[47] Although PSV categorization and performance requirements were not explicitly addressed, the paper does include an issue on "Definition of Passive Failure" and an extensive discussion on whether check valves are passive or active components and how they should be addressed in current plants and future passive designs.

SECY-94-084 recognizes the GDCs and SECY-77-439 as establishing long-standing requirements and guidance in this area. The paper acknowledges that the industry (including EPRI documents and ANSI/ANS 58.9[48]) have been inconsistent with respect to check valve failures, sometimes considering them as "active failures" and sometimes as "passive failures." In SECY-77-439, however, the NRC staff stated that the failure of a simple check valve to move to its correct position when required was a "passive failure." In addition, SECY-94-084 states that "[i]n licensing reviews, however, only on a long-term basis [e.g., long-term recirculation cooling following a loss of coolant accident (LOCA)] does the NRC staff consider passive failures in fluid systems as potential accident initiators in addition to initiating events." The paper also states that "[f]or current plants, the NRC staff normally treats check valves, except for those in containment isolation systems, as passive devices during transients or design-basis accidents."

Furthermore, SECY-94-084 states that "[r]edefining check valves as active components, subject to consideration for single active failures would cause these valves to be evaluated in a more stringent manner than that used in previous licensing reviews" (emphasis added). The NRC staff then recommended (and the Commission agreed[49]) that the NRC staff should "maintain the current licensing practice for passive component failures on the passive [advanced light water reactor] ALWR designs, and to redefine check valves, except for those whose proper function can be demonstrated and documented, in the passive safety systems as active components subject to single failure consideration." Therefore, the NRC's position on check valves was changed only for passive ALWR designs going forward.

The Panel considers the opening function of check valves and PSVs to be similar in that they both open through the motion of the valve disk under differential pressure with no external signal or motive power. The Panel also recognizes that the ambiguity with respect to "passive" versus "active" component definitions and nomenclature exists for safety valves. In addition, the passive or active classification of check valves or safety valves may differ based on design considerations, inservice testing, or accident analyses. For example, the PSVs and PORVs as well as numerous check valves are classified as active components in the Byron and Braidwood inservice testing programs. However, for purposes of applying the single failure criterion in the GDC context, the Panel concludes that it is appropriate to consider the potential failure of a PSV following water discharge as a passive failure, consistent with the treatment of check valve failures for the operating fleet.

3.7 Draft Standard Review Plan Revision (1996)

The 1996 draft revision to SRP Section 15.5.1 – 15.5.2 on IOECCS and CVCS malfunctions includes extensive updates to the 1981 revision, but neither version includes any discussion, criteria, or guidance on applying ASME Code requirements to PSVs or on applying the single failure criterion or any other failure assumption to PSVs.[50]

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[47] NRC 1994a
[48] ANS 1981
[49] NRC 1994b
[50] NRC 1996

3.8 Power Uprate Reviews and License Amendments (2001-2006)

As part of the 2001 power uprate review for Byron and Braidwood, the NRC staff approved the analysis of an IOECCS (UFSAR Section 15.5.1) that included pressurizer filling, PSV water discharge, ECCS termination, and PSV closure. In the Backfit SE, the NRC staff indicates that the 2001 license amendment was predicated on the NRC's mistaken (unsubstantiated) belief that the valves were ASME-qualified (certified). However, a review of the SE and associated RAIs shows that, in 2001, the NRC staff was well aware of the nature of the EPRI testing that the licensee relied on. The Panel did not find any evidence that the licensee claimed or the NRC staff believed that the valves were "qualified" in an ASME certification sense; rather, the record shows thorough consideration of the testing conducted on valves of the type installed at the plant and a technical judgment that this testing provided appropriate qualification.

The Panel's conclusion was confirmed via discussions with the individual who was the responsible Section Chief in the Reactor Systems Branch at the time. He informed the Panel that the 2001 license amendment was based on the exercise of staff engineering judgment and there was no discussion of ASME certification or qualification of valves. In addition, the Panel found that the NRC approved power uprates for other nuclear power plants that included staff evaluation of water discharge through PORVs or PSVs based on test information provided by individual licensees. For example, in 2001, the NRC granted a power uprate for Shearon Harris that included the operability of PORVs and PSVs during the discharge of subcooled water, referencing TMI Action Plan Item II.D.1.[51] As noted above, in 2006, the NRC also granted a power uprate for Beaver Valley. The SE for this Beaver Valley amendment referred to RIS 2005-29 and found reasonable assurance that the PSVs would adequately discharge water and reseat following a spurious safety injection actuation, based on the EPRI test data from 1981 and an evaluation of the temperature of the liquid being discharged.

During the NRC evaluations of license amendments since the TMI-2 accident, the NRC staff has specified in some SEs that a PORV or PSV would be assumed to stick open if it was not qualified for liquid service. To address this concern, the NRC staff reviewed and accepted a variety of test information (including EPRI, Wyle, and vendor testing) submitted by individual licensees to demonstrate the capability of PORVs or PSVs to reseat following water discharge. In the sample of SEs it reviewed, the Panel did not find a specific requirement for the PORVs or PSVs to be certified under the ASME BPV Code as capable of passing water and reclosing.

In 2004, the NRC issued license amendments for Byron and Braidwood granting an adjustment to the PSV setpoints. In an RAI, the NRC staff requested that the licensee perform a quantitative analysis regarding the number of opening cycles during which the PSV would be expected to pass water and the temperature of the water being discharged. In the Setpoint SE, the NRC staff concluded that the analysis was acceptable for assuring that the PSVs would remain operable following a spurious safety injection event.

[51] NRC 2001d

3.9 RIS 2005-29 (2005)

In 2005, the NRC staff issued RIS 2005-29 "to notify licensees of a concern identified during recent reviews of power uprate [LARs]." The RIS addressed the manner in which some licensees acted in response to NSAL-93-013. The RIS was issued at the division level in NRR and does not include a record of office-level concurrence. The RIS was not reviewed by CRGR. Although no documentation was readily available regarding the CRGR's decision not to review, it appears that the lack of a CRGR review stemmed from the assertions in the RIS such as these:

- "This RIS requires no action or written response and, therefore, is not a backfit under 10 CFR 50.109. Consequently, the NRC staff did not perform a backfit analysis."
- "This RIS is informational and pertains to a NRC staff position that does not depart from current regulatory requirements and practice."

A key statement in RIS 2005-29 is the following (with emphasis added):

> The NRC staff's position is noted in the power uprate review standard, as follows: "For the [IOECCS] and [CVCS] malfunctions that increase reactor coolant inventory events: (a) non-safety-grade pressure-operated relief valves should not be credited for event mitigation and (b) <u>pressurizer level should not be allowed to reach a pressurizer water-solid condition.</u>".

However, the cited review standard (RS-001), which is explicitly limited to EPUs, states that "[t]he staff does not intend to impose the criteria and/or guidance in this review standard on plants whose design bases do not include these criteria and/or guidance. No backfitting is intended or approved in connection with the issuance of this review standard."[52]

This intent of RS-001 to define and clarify the scope of EPU reviews, but not impose new requirements or new interpretations of requirements, was confirmed by the Panel in discussions with the manager responsible for developing and issuing RS-001. Therefore, contrary to the RIS statement, neither RS-001 nor RIS 2005-29 documented "known and established standards of the Commission" applicable to Byron and Braidwood.

The Panel also notes that neither RIS 2005-29 nor its draft Revision 1,[53] which is currently under development, discuss water discharge certification requirements in accordance with the ASME BPV Code. In fact, as stated above, the NRC issued a 2006 power uprate amendment for Beaver Valley in which the SE cited RIS 2005-29 and yet relied on the EPRI testing data to address the concern.

3.10 SECY-05-0138 (2005)

SECY-05-0138 presents a comprehensive history of the application of the single failure criterion, including extensive discussion of the treatment of passive components in fluid systems.[54] The paper enclosed a July 2005 draft of a technical report on the single failure criterion. Section 4.2.2 of this report acknowledges that "[o]ne particular issue identified in this

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[52] NRC 2003
[53] NRC 2015a
[54] NRC 2005a

project is the continued existence of the footnote to the definition of single failure in 10 CFR [Part] 50 Appendix A stating that the regulatory position on considering passive failures in fluid systems is under development." In Section 2.5.3, the draft report quotes from SECY-77-439 (discussed above) and recognizes that in current practice, as in 1977, "[p]assive failures in fluid systems are generally excluded from single-failure assessments."

SECY-05-0138 and the accompanying draft report present three alternatives for using a risk-informed and performance-based approach to address the single failure issue. The draft report clarifies that all of the alternatives "could include developing a position on single passive failures in fluid systems to replace the footnote now in 10 CFR Part 50 Appendix A definitions."

These documents make it clear that, with few exceptions, neither the NRC staff nor the Commission has established specific requirements relating to the treatment of passive component failures in fluid systems. The Panel believes the existence of this Commission paper, contemporaneous with discussions on potential PSV failures (e.g., RIS 2005-29), makes it clear that no specific "known and established standards" on PSV failures had been developed between 1977 and the time of the Byron and Braidwood license amendments in 2001 and 2004.

3.11 Standard Review Plan Revision (2007)

Revision 2 to SRP Section 15.5.1 – 15.5.2 states:

> If the plant is equipped with PORVs that are (1) safety-related equipment and (2) qualified for water relief, then they may be assumed to reseat properly after having relieved water. The [PSVs], too, may be assumed to reseat properly after having relieved water; but only if such valves have been qualified for water relief.

However, this section does not reference ASME BPV Code requirements for safety valve certification.

3.12 Backfit Letter and Subsequent Appeals (2015-2016)

The Backfit SE is predicated on the following positions:

- "water relief through a valve that is not qualified for water relief <u>will cause</u> that valve to stick in its <u>fully open position</u>" (emphasis added)
- "the licensee … has not applied the <u>single-failure assumption</u>" (emphasis added)
- "nor have they provided <u>ASME water qualification</u> documentation for the PSVs … the ASME … original Overpressure Protection Report … inservice test history … including both water and steam tests" (emphasis added)

The Backfit SE argues that an IOECCS would escalate to a more severe event. Such an escalation would be contrary to the Byron and Braidwood licensing basis (i.e., contrary to the ANS non-escalation position) and could be in non-compliance with the GDCs (as included in the Byron and Braidwood licensing basis) since an IOECCS with a stuck-open valve had not been analyzed and shown to meet the appropriate criteria for an AOO.

Based on its review of all the relevant documents and discussions with the individuals (staff and managers) involved in the original review and the backfit, the Panel has developed an understanding of the regulatory requirements and practices, the potential safety issues, and backfit rule obligations. The Panel has determined that the numerous, complex, and detailed regulatory and technical issues all depend on the answers to two critical questions on valve performance:

- Must the PSVs in question be assumed to fail given liquid water discharge because of the lack of ASME certification for water discharge?
- Must the PSVs be assumed to fail in accordance with the GDC "single failure" requirements?

In the Backfit SE, the NRC staff indicates that "[o]ne assumption that is particularly important to the non-escalation criteria is that water relief through a valve that is not qualified for water relief will cause that valve to stick in its fully open position" (emphasis added). The Panel concludes that this issue—the treatment of potential valve failure—is not only "particularly important," it is the critical issue upon which the compliance backfit hinges.

Based on the historical evidence, the Panel concludes that there is not now, nor has there been, a known and established Commission standard (1) that PSVs must be assumed to fail following water discharge in the absence of ASME certification for water discharge, or (2) that PSVs must be assumed to fail as part of single failure criterion analysis. The NRC staff's determination that ASME certification is necessary first appears in the Backfit SE. The determination that application of the single failure criterion is necessary first appears in the draft Revision 1 to RIS 2005-29. The Panel has not identified these positions being stated in any final NRC guidance document.

The Panel also concludes that in 2001 and 2004 and at present, the known and established standard of the Commission is 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. In preparing the Uprate SE and the Setpoint SE, the NRC staff exercised reasonable and well-informed engineering judgment when the NRC staff concluded that the PSVs were unlikely to stick open. The NRC staff reviewers involved in the 2001 power uprate review were among the most experienced and senior reviewers in their areas of expertise. The NRC staff valve expert involved in the review was the agency's most knowledgeable individual on PSVs and the relevant ASME Code requirements, and was a nationally recognized expert. The Panel did not find any evidence that the NRC staff's issuance of the 2001 or 2004 license amendments was based on an omission or mistake of fact. Rather, the current NRC staff positions on valve qualification in the Backfit SE are new or modified interpretations of compliance.

In interactions with the Panel, NRR staff emphasized several issues raised in the Backfit Letter. The Panel summarizes its consideration of those issues in the following subsections.

3.12.1 Non-Escalation Position and Valve Failure

In the Backfit SE, the NRC staff discussed the definition of event conditions in ANS-51.1/N18.2-1973 and the provision in this standard that events of one condition do not propagate to cause a more serious fault (non-escalation position). In interactions with the Panel, NRR staff provided several clarifications on this topic, summarized by the Panel as follows:

- ANS-51.1/N18.2-1973 defines the categories of design basis transients and accidents based on an anticipated frequency of occurrence (annually for ANS Condition II events).
- It is a long-standing NRC position that escalation from one condition to another is not acceptable.
- ANS-51.1/N18.2-1973 constitutes a known and established standard that has been reflected in NRC guidance documents and in the licensing basis of each U.S. nuclear power plant.

The Panel confirmed that this ANS standard is referenced in several places in Chapter 15 of the Byron and Braidwood UFSAR. The Panel agrees that the non-escalation position is an established standard applicable to Byron and Braidwood, but did not identify historical evidence that implementation of this standard requires Exelon to assume that its pressurizer valves will fail open under water discharge conditions, to apply the single failure criterion to PSV failure in these circumstances, or to impose ASME Code requirements for certification, qualification, or testing of PSVs for water discharge.

3.12.2 Non-Escalation Position and Return to Service

In the Backfit SE, the NRC staff makes reference to the time it would take to clean up a contaminated containment following a stuck-open pressurizer valve. In interactions with the Panel, NRR staff re-emphasized concerns that extended steam and water discharge through the pressurizer valves would result in the failure of the pressurizer relief tank rupture disk, would require repair of the damaged PSVs, and might cause an extended time period for the return to service of the nuclear power plant.

The Panel does not consider the time period necessary for the licensee to perform radioactive clean-up activities in the containment building, to inspect and conduct any necessary repairs to the PSVs, or to prepare for plant startup, to constitute issues that support a compliance backfit imposed by the NRC. The NRC staff and inspectors would verify that these activities are conducted appropriately to protect the public health and safety prior to plant restart. The Backfit SE states that UFSAR Section 15.5.1.3 "implie[s]" that the plant will return to operation in a "short period," but the Panel sees no support for a timing requirement in UFSAR Section 15.5.1.3. Also, the Panel has not identified a regulatory interest in limiting the time needed for the plant to return to operation.

3.12.3 TMI Action Plan Item II.D.1 and EPRI Testing

Although the Backfit Letter and NRR Appeal Decision do not speak explicitly to TMI Action Plan Item II.D.I, in interactions with the Panel, NRR staff stated that the known and established standard in question is the TMI Action Plan Item II.D.I standard for licensees and applicants to conduct testing to qualify the RCS relief and safety valves under expected operating conditions for design-basis transients and accidents. As discussed above and in Appendix B to this report, the NRC accepted the EPRI testing to satisfy TMI Action Plan Item II.D.1 for Byron and Braidwood in SEs forwarded by letters in 1988 and 1990. Therefore, the Panel considers this known and established standard referenced by the NRC staff to have been met for Byron and Braidwood.

In interactions with the Panel, the NRR staff further stated that an omission or mistake of fact occurred when the licensee failed to acknowledge that the EPRI testing program did not evaluate water discharge from the pressurizer valves during extended high pressure safety injection for Byron and Braidwood. As discussed in Appendix B to this report, in the 1988 and 1990 SEs on the Byron and Braidwood response to TMI Action Plan Item II.D.1, the NRC staff evaluated the capability of the PSVs and PORVs during feedwater line break accidents, including water discharge. In these SEs, the NRC staff found that the performance of the PSVs and PORVs with water discharge was acceptable based on the EPRI tests. Therefore, the Panel does not agree that the licensee's reference to the EPRI testing program was an omission or mistake of fact.

3.12.4 ASME Code Certification

In the Backfit SE, the NRC staff stated that certain ASME Code information would be necessary to support water qualification of the PSVs. In interactions with the Panel, NRR staff stated that, to satisfy the standard for water discharge capability of pressurizer valves, it would be necessary to conduct flow capacity certification in accordance with the ASME BPV Code and inservice testing throughout the service life in accordance with the ASME OM Code. The NRR staff referenced certain licensing actions in which water discharge was not considered acceptable, or different actions were required.[55]

As discussed in Appendix C to this report, the NRC staff required additional actions for some licensees to support reliance on the PORVs for water discharge and to avoid water discharge through the PSVs. The Panel found, however, that the NRC staff also allowed some licensees to rely only on EPRI testing without significant additional activities. The Panel did not identify instances where the NRC staff imposed certification by the ASME BPV Code and testing in accordance with the OM Code, or required alternatives to the ASME BPV or OM Codes, in the examples of NRC staff review of water discharge capability for pressurizer valves.

In interactions with the Panel, the NRR staff also identified specific ASME Code provisions that it viewed as supporting the position that ASME Code requirements apply to qualification of pressurizer valves for water discharge. The NRR staff, however, did not provide evidence that these provisions have consistently been interpreted as the NRC staff is now interpreting them. Given the NRC's treatment of TMI Action Plan Item II.D.1 and the NRC staff's historical licensing practice, the Panel concludes that the NRR staff's current application of the ASME Code is not supported by the historical record.

3.12.5 Conduct of 2001 and 2004 Reviews

In light of the wide range of NRC staff positions during the review of pressurizer valve capability since the TMI-2 accident, the Panel agrees that, in the course of preparing the 2001 Uprate SE or Setpoint SE, the NRC staff could have considered the need for the licensee for Byron and Braidwood to improve the reliability of the PSVs or PORVs for water discharge or to avoid water discharge through the PSVs by PORV improvements. The NRC staff may have been able to justify additional actions, but they determined that it was not necessary. Instead, the NRC staff reviewers in 2001 used their expert engineering judgement to determine that it was not necessary to assume that the PSVs or PORVs would stick open with water discharge, based on EPRI test information, licensee supplemental information, and their own technical experience.

In discussions with the Panel, NRR staff raised a concern that the Setpoint SE does not document a re-review of the qualification of the PSVs and noted that if the Uprate SE had not found water discharge through the PSVs to be acceptable, it is unlikely that the NRC staff would

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[55] Salem (NRC 1997), Millstone (NRC 1998), and Callaway (NRC 2000)

have approved this 2004 amendment. In Appendix C to this report, the Panel summarizes the discussion in the Setpoint SE of the PSV water discharge capability. The Panel recognizes that a staff review may rely on a previous more extensive review to determine the acceptability of a similar request. The Panel does not consider the review approach used in 2004 to challenge the adequacy of the 2001 review.

4 RESPONSE TO THE EDO QUESTIONS

In establishing the Panel, the EDO asked the Panel to answer five specific questions, as well as evaluating the overall appropriateness of the backfit. The answers to these questions are provided below.

4.1 Were the approvals based on a mistake? If so, what was the mistake and what are the implications for Braidwood and Byron?

In responding the question, the Panel has considered the differing views of the NRR staff and the licensee on this issue. Those positions are summarized below:

- In the NRR Appeal Decision, the NRC staff claims that "[t]he NRC erred in approving a sequence of events that allowed the [IOECCS], [CVCS] malfunction, and inadvertent opening of a pressurizer safety or relief valve analyses in the 2001 and 2004 [SEs]" and "the NRC staff understood the PSVs to be qualified for water relief when, in fact, they were not."

- Exelon claims in the NRR Backfit Appeal that "the compliance exception requires more than simply asserting that the prior staff approvals were wrong—the NRC must demonstrate that the prior approvals were erroneous because of an omission or mistake of fact at the time of the approval. The NRC has not made that case here."

The Panel concludes that, in 2001 and 2004, the NRC staff did <u>not</u> misunderstand the qualification status of the PSVs and that it was <u>not</u> a mistake to undertake a review of or make a technically based safety finding on the likely successful performance of the valves. In the Panel's opinion, the actions of the Reactor Systems Branch in 2001 to reach out to the Division of Engineering's Mechanical Engineering Branch for expert technical review assistance was both appropriate and commendable. The NRC staff reviewers involved in the 2001 power uprate review were among the most experienced and senior reviewers in their areas of expertise. The valve expert involved in the review was the NRC's most knowledgeable individual on PSVs and the relevant ASME Code requirements, and was a nationally recognized expert. The Panel cannot agree that the NRC staff was misinformed, ill-informed, or in error, or that it made incorrect or inappropriate decisions.

4.2 What is the known and established standard for water qualification of PSVs?

The Panel concludes 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. No more detailed or prescriptive standard has been promulgated by the Commission.

4.3 What is the known and established standard for progression of postulated events between categories of severity?

For Byron and Braidwood, the standard for progression of postulated events between categories of severity is set forth in the UFSAR as described above. The Panel supports the NRC staff's view that non-escalation (from ANS Condition II to ANS Condition III or IV) is a known and established standard applicable to Byron and Braidwood. This issue of event escalation is also a focus of RIS 2005-29 and the draft Revision 1 to RIS 2005-29 that was issued for public comment in 2015. The Panel concludes that the IOECCS (an AOO per the GDC definition and an ANS Condition II event) would escalate to a more severe event if a PSV were to stick open, or if both a PORV stuck open and its block valve failed to close. Such an escalation would be contrary to the Byron and Braidwood licensing basis (i.e., contrary to the ANS non-escalation position) and could be in non-compliance with the GDC (as included in the Byron and Braidwood licensing basis), since an IOECCS with a stuck-open valve had not been analyzed and shown to meet the appropriate criteria for an AOO. However, this event progression standard does not establish specific standards for valve qualification to determine whether a valve would stick open and cause this escalation. Therefore, it is not the basis for a compliance backfit given the current set of facts.

4.4 Does the current licensing basis for Braidwood and Byron comply with the applicable regulations? Is it adequate to provide protection to public health and safety?

The Panel concludes that Byron and Braidwood do comply with the applicable regulations based on the UFSAR analyses, which the NRC staff found acceptable through a reasonable and technically sound evaluation using appropriate Commission safety standards.

4.5 Given that Exelon suggests that the NRC pursue a cost-justified substantial safety enhancement backfit, what is the contribution to overall plant risk of the current configuration at Braidwood and Byron?

The Panel requested RES to provide information and insights on the risk significance of the sequence at issue, to assure that the Panel's judgments were being made with a full understanding of their significance, and to assist in responding to the EDO question.

The RES study suggests that the most significant IOECCS sequence, assuming that all pressurizer overfill events lead to a small LOCA, contributes approximately 1 percent of the total internal event core damage frequency (CDF). In its report, RES estimated a maximum benefit (CDF reduction) from a "perfect backfit" (i.e., always preventing pressurizer overfill) of 1.5E-07 per year. If the PSVs are not assumed to always fail following water discharge (consistent with the NRC staff expert judgment in 2001) or a smaller improvement than a "perfect backfit" were considered, the risk-reduction benefit of implementing the backfit would be even smaller.

The Panel is aware of and sensitive to two important issues related to this question. First, NRR, not the Panel, is responsible for any decisions on alternative application of the backfit rule to this issue (through the other categories of adequate protection or cost-justified substantial safety enhancement). Second, the Panel does not wish to imply that "the contribution to plant risk" should be seen as the only measure of enhanced safety. The issues of event classification and the non-escalation of events are essentially defense-in-depth concepts. Defense in depth has a recognized role and value in the regulatory process. The Panel is also aware that not every defense-in-depth feature has the same safety significance, and that the estimated risk significance (measured in core damage frequency) is very relevant.

Within the context described above, the Panel concludes that the contribution to overall plant risk is very small.

5 SUMMARY AND CONCLUSIONS

The compliance exception to the Backfit Rule is intended to address failures to meet known and established Commission standards because of omission or mistake of fact. New or modified interpretations of what constitutes compliance do not fall within the exception. Therefore, to address the appeal of the proposed compliance backfit, the Panel focused on determining if this case is most appropriately characterized as one in which the licensee "failed to meet known and established standards of the Commission because of omission or mistake of fact," or rather as a case of a "new or modified interpretations of what constitutes compliance."

The NRC staff's compliance backfit argument depends on two separate determinations:

1. the assumed failure of PSVs to reclose after passing water, and
2. the necessity of preventing "event escalation" (i.e., the position that "an incident of moderate frequency should not generate a more serious plant condition without other faults occurring independently").

For the NRC staff's compliance backfit conclusion to be valid, both of these determinations must meet the above compliance backfit standard by involving failure to meet known and established standards of the Commission.

In the first of these determinations, the NRC staff's compliance backfit is based on the assumption in the Backfit SE that the PSV fails to reclose given the absence of "ASME water qualification documentation." As indicated in the Backfit SE, the Uprate SE involved a technical evaluation of safety valve capability and likely performance under water-discharge conditions rather than a simple assumption of a failure. The NRR Appeal Decision indicates that "the 2001 and 2004 approvals occurred because the NRC staff understood the PSVs to be qualified for water relief when, in fact, they were not."

The Panel carefully considered these views and has reviewed the relevant documents including the licensee's responses to the NRC staff's RAIs,[56] the technical branch's SE input,[57] and the Uprate SE. 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 certification sense; rather, the record shows thorough consideration of the testing conducted on valves of the type installed at the plant and a technical judgment that this testing provided appropriate qualification.

On the basis of its review, the Panel concluded that the NRC staff who prepared the Uprate SE did <u>not</u> misunderstand the qualification status of the PSVs and that it was <u>not</u> a mistake to undertake a review of or make a technically based safety finding on the likely successful performance of the valves. In the Panel's opinion, the actions of the Reactor Systems Branch in 2001 to reach out to the Division of Engineering's Mechanical Engineering Branch for expert

[56] ComEd 2000b, Exelon 2001
[57] NRC 2001a

technical review assistance was both appropriate and commendable. The NRC staff reviewers involved in the 2001 power uprate review were among the most experienced and senior reviewers in their areas of expertise. The NRC staff valve expert involved in the review was the agency's most knowledgeable individual on PSVs and the relevant ASME Code requirements, and was a nationally recognized expert. The Panel disagreed that the NRC staff was misinformed, ill-informed, or in error, or that it made incorrect or inappropriate decisions.

The Panel concluded that three related technical and regulatory positions related to the PSVs (separate from the issue of the non-escalation position) underpin the backfit:

1. ASME water qualification (certification) documentation is required if a valve is to be assumed to reclose after passing water.

2. Water discharge through a steam-qualified valve will cause that valve to stick in its fully open position.

3. PSVs are subject to a single-failure assumption.

None of these positions were "known and established standards of the Commission" in 2001 or 2004 for determining when it was appropriate to assume a failure of PSVs to reseat. In fact, they were not "known and established standards of the Commission" in 2005 (when RIS 2005-29 was issued) or 2006 (when the Beaver Valley EPU was approved) or 2007 (when Revision 2 to SRP Section 15.5.1 – 15.5.2 was issued).

Moreover, these positions do not appear to be "established standards of the Commission" at present. The 2007 version of SRP Section 15.5.1 – 15.5.2 allows credit for PORVs and PSVs if they have been "qualified for water relief." The NRC staff's determination that ASME certification is necessary first appears in the Backfit SE and is not addressed in any final NRC guidance document. The determination that application of the single failure criterion is necessary first appears in the draft Revision 1 to RIS 2005-29 and is not included in any final NRC guidance document.

The Panel 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. In earlier documents addressing this topic, beginning with NUREG-0737, the use of the word "qualified" or "qualification" implies a general demonstration of capability, such as in the EPRI testing done in response to TMI Action Plan Item II.D.1. In light of this standard, the Panel concluded that, when preparing the Uprate SE and the Setpoint SE, the NRC staff exercised reasonable and well-informed engineering judgment to conclude that the PSVs were unlikely to stick open.

Overall, the Panel concluded that the NRC staff's position on valve qualification in the Backfit SE is a new or modified interpretation of what constitutes compliance in addressing potential PSV failures following water discharge. Although this new staff position represents a well-intentioned and conservative approach that could provide additional safety margin, it does not provide a basis for a compliance backfit.

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.

The Panel's findings, therefore, support the Exelon backfit appeal.

6 ADDITIONAL PANEL THOUGHTS

In addition to the specific finding relating to the backfit appeal, the Panel believes it is important to acknowledge, and for the NRC staff and licensees to appreciate, that water discharge through a PSV not specifically designed for such service is undesirable and should be minimized or avoided as a matter of conservative engineering and prudent operations. This is reinforced by the information provided in NSAL-93-013 and its Supplement 1, and the actions by various licensees in response to these documents, as well as the limited scope of the EPRI testing conducted over 30 years ago.

Operator training, control room procedures to terminate the event before pressurizer filling, and use of PORVs rather than reliance on PSVs, are clearly preferred and prudent measures, whether they form the facilities' UFSAR licensing basis and are assumed in the accident analyses or not.

The PSVs in question were designed for steam service. Steam relief is their normal service condition and applies to their ASME BPV Code certification. The Panel supports the previous NRC staff determinations for Byron and Braidwood and certain other plants that PSVs experiencing water discharge during an abnormal or accident condition need not be assumed to fail since there was a reasonable and technically well-informed engineering judgement to the contrary. However, the Panel also considers the actions by various licensees to improve the reliability and performance of the PORVs to avoid water discharge through the PSVs to be prudent in light of the design specifications of the PSVs.

The Panel considered but could not determine the extent to which the licensee for Byron and Braidwood addressed crediting water discharge through the PSVs, PORVs, or PORV block valves in the Byron and Braidwood inservice testing programs. The Panel recognizes that the difference between the intended use of these valves for overpressure protection and their infrequent use in response to certain plant events might be considered in implementing appropriate inservice testing activities.

The Panel notes that water discharge through various pressurizer valves is not a new issue because water discharge has always been credited (by the licensee for Byron and Braidwood and other licensees) for the feedwater line break analysis in UFSAR Section 15.2.8.

APPENDIX A: HISTORY OF THE BACKFIT RULE AND THE COMPLIANCE EXCEPTION

The Backfit Rule

Title 10 of the *Code of Federal Regulations* (10 CFR), Section 50.109, "Backfitting," was originally promulgated in 1970.[58] Because of perceived deficiencies in the rule, the U.S. Nuclear Regulatory Commission (NRC) substantially revised it in 1985.[59] The 1985 rule was challenged in court, and the U.S. Circuit Court for the District of Columbia (D.C. Circuit) vacated this rule in its entirety. The D.C. Circuit took this action because it concluded that the revised rule could be interpreted to allow the NRC to consider costs in defining or redefining what is required for adequate protection of the public health and safety.[60] In response, the NRC revised the Backfit Rule in 1988 to remove any implication that costs could be considered in defining or redefining adequate protection.[61] The 1988 revisions only differed from the 1985 rule to the extent necessary to address the court's concerns. The 1988 rule was also challenged in court, but this time the D.C. Circuit upheld the rule.[62]

In its current form, 10 CFR 50.109(a)(1) defines backfitting as

> … the modification of or addition to systems, structures, components, or design of a facility; or the design approval or manufacturing license for a facility; or the procedures or organization required to design, construct or operate a facility; any of which may result from a new or amended provision in the Commission's regulations or the imposition of a regulatory staff position interpreting the Commission's regulations that is either new or different from a previously applicable staff position … .

Unless one of three specified exceptions apply, the NRC may impose a backfit only if it performs a backfit analysis in accordance with 10 CFR 50.109(a)(2) and determines in accordance with 10 CFR 50.109(a)(3) "that there is a substantial increase in the overall protection of the public health and safety or the common defense and security to be derived from the backfit and that the direct and indirect costs of implementation for that facility are justified in view of this increased protection."

Section 50.109(a)(4) sets forth the three exceptions to the requirements of 10 CFR 50.109(a)(2) and (a)(3). The first exception, the compliance exception, applies if the "modification is necessary to bring a facility into compliance with a license or the rules or orders of the Commission, or into conformance with written commitments by the licensee." 10 CFR 50.109(a)(4)(i). The second and third exceptions relate to actions ensuring adequate protection or to actions that involve defining or redefining adequate protection. 10 CFR 50.109(a)(4)(ii)-(iii).

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[58] AEC 1970 (Author and year citations in footnotes refer to the designation of references in Appendix D to this report.)
[59] NRC 1985
[60] *Union of Concerned Scientists v. U.S. Nuclear Regulatory Com'n*, 824 F.2d 108, 119-20 (1987).
[61] NRC 1988b
[62] *Union of Concerned Scientists v. U.S. Nuclear Regulatory Com'n*, 880 F.2d 552 (1989).

Commission Policy

The Commission addressed its intended application of the compliance exception in the 1985 rulemaking:[63]

> The compliance exception is intended to address situations in which the licensee has failed to meet known and established standards of the Commission because of omission or mistake of fact. It should be noted that new or modified interpretations of what constitutes compliance would not fall within the exception and would require a backfit analysis and application of the standard.

In the 1985 rule, the Commission acknowledged that staff interpretations of regulations are not legally binding, but the Commission also stated that "staff interpretations of broadly stated rules are often necessary to give a rule effect and in some instances may be a causal factor in initiating a backfit."[64] The Commission also stated, "Many of the most important changes in plant design, construction, operation, organization, and training have been put in place at a level of detail that is expressed in staff guidance documents which interpret the intent of broad, generally worked [sic] regulations."[65]

Backfitting Guidance

Extensive information regarding the appropriate implementation of backfitting is provided in NUREG-1409.[66] Relevant excerpts from this guidance are provided below.

Applicable Regulatory Staff Positions

According to NUREG-1409, to be a backfit, "a new or revised staff position or requirement must be involved, that is, there must be a change in content or applicability of the previously applicable regulatory staff position (in the direction of increased safety requirements) … ." An applicable regulatory staff position is a requirement or position already specifically imposed on or committed to by a licensee. Examples of applicable regulatory staff positions include:

- legal requirements, as in explicit regulations, orders, and plant licenses and in amendments, conditions, and technical specifications
- written licensee commitments such as those contained in the final safety analysis report, licensee event reports, and docketed correspondence, including responses to NRC bulletins, generic letters, inspection reports, or notices of violation and confirmatory action letters
- NRC staff positions that are documented explicit interpretations of more general regulations and are contained in documents such as the Standard Review Plan, branch technical positions, regulatory guides, generic letters, and bulletins

A similar list of examples is provided in Manual Chapter 0514,[67] which is also included as Appendix D to NUREG-1409. Manual Chapter 0514 was referenced in the 1988 rulemaking,

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63 NRC 1985, at 38103
64 *Id.* at 38102
65 *Id.* at 38103. The 1988 rulemaking neither revised the compliance exception as stated in the 1985 rule nor provided additional guidance on its interpretation.
66 NRC 1990c

and a working draft was provided to the Commission for information in SECY-88-102.[68] Manual Chapter 0514 provides a definition of "applicable regulatory staff positions" that is slightly more detailed than the definition in NUREG-1409. This definition from Manual Chapter 0514 is quoted below, with additional detail beyond NUREG-1409 emphasized in underlined text.

> Applicable regulatory staff positions are those already specifically imposed upon or committed to by a licensee at the time of the identification of a plant-specific backfit, and are of several different types and sources:
>
> a. Legal requirements such as in explicit regulations, orders, plant licenses (amendments, conditions, technical specifications). Note that some regulations have update features built in, as for example, 10 CFR 50.55a, Codes and Standards. Such update requirements are applicable as described in the regulation.
>
> b. Written commitments such as contained in the [Final Safety Analysis Report], [Licensee Event Reports], and docketed correspondence, including responses to Bulletins, responses to Generic Letters, Confirmatory Action Letters, responses to Inspection Reports, or responses to Notices of Violation.
>
> c. NRC staff positions[69] that are documented, approved, explicit interpretations of the more general regulations, and are contained in documents such as the [Standard Review Plan], Branch Technical Positions, Regulatory Guides, Generic Letters, and Bulletins; and to which a licensee or an applicant has previously committed to or relied upon. Positions contained in these documents are not considered applicable staff positions to the extent that staff has, in a previous licensing or inspection action, tacitly or explicitly excepted the licensee from part or all of the position.[70]

How Regulatory Positions are Established

NUREG-1409 provides responses to a number of questions regarding backfitting. The following response was given to questions asking, "Is it appropriate for the NRC staff to rely on informal or formal communications to other licensees as official NRC positions? What about NRC tacit approval of documents?"

> Informal or formal communications to one licensee are not official positions to all licensees. Section 053 of Manual Chapter 0514 identifies what can be applied as official staff positions in a plant-specific context. They are legal requirements such as contained in explicit regulations, orders, and plant licenses; written commitments such as contained in final safety analysis reports, licenses event reports, and docketed correspondence; and documented, approved explicit interpretations such as contained in the [Standard Review Plan], branch technical positions, regulatory guides, generic letters, and bulletins. Orders, licenses, and written commitments are applicable only to a particular licensee.

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[67] NRC 1988c
[68] NRC 1988a
[69] Requirements may be imposed by rule or order. Staff interpretations such as examples of acceptable ways to meet requirements are not requirements in and of themselves.
[70] Imposition of a staff position from which a licensee has previously been excepted is a backfit.

If the NRC staff previously exempted a licensee from a legal requirement or approved position, it is not applicable to that licensee for the purpose of backfit consideration. Explicit exemption would be done formally in writing. The Appendix to NRC Manual Chapter 0514 discusses tacit approval under reanalysis of issues. Two situations are covered. In the first case, staff review of a previously accepted licensee action or program may result in a requested change. This would be classified as a backfit because it represents a change in a previous staff position and would require a backfit analysis (or a documented evaluation if it meets one of the exceptions listed in the backfit rule). In the second case, a licensee submittal committing to a specific course of action that has not received timely NRC staff review is implemented by the licensee. In this case, it is considered that the NRC staff tacitly accepted the licensee's action since timely notice to the contrary was not given. If the NRC staff subsequently adopts a different position and requests a change in the licensee action, this change may be classified as a backfit and thus require a backfit analysis (or a documented evaluation if it meets one of the exceptions listed in the backfit rule).

NUREG-1409 also addresses a question regarding tacit approvals by an inspector: "If an inspector has previously accepted (i.e., provided tacit approval of) a licensee's method, does a specific request for change constitute a backfit and if so, is a backfit analysis required?" The response is:

> Cases where an inspector provides tacit approval are relatively rare. Simply not challenging a licensee's practice normally would not be considered tacit approval. The only example provided in Manual Chapter 0514 is a case where the NRC has indicated tacit approval by not acting in a reasonable time on a licensee submittal and the licensee has moved ahead to implement the proposal described in the submittal. For the purpose of this question, it would most likely arise in connection with review of a licensee response to an inspection report.
>
> Explicit approval could be provided in an inspection report that states that a particular approach is acceptable. However, conclusions of that nature are usually made in [safety evaluations] rather than inspection reports.

Compliance Backfit Guidance

NUREG-1409 gives the following response to the question, "[h]ow does the backfit rule apply to new staff positions that reflect an evolving understanding of technical issues?"

> An evolving understanding of issues does not, by itself, define which category fits a particular backfit. Judgment must be applied to the facts of each particular case to determine whether the backfit is for compliance, to provide adequate protection, to redefine adequate protection, or to achieve a cost-justified substantial safety enhancement. For example, with regard to compliance, the 1985 statement of considerations for 10 CFR 50.109 indicates that "the compliance exception is intended to address situations where the licensee has failed to meet known and established standards of the Commission because of omission or mistake of fact....new or modified interpretations of what constitutes compliance would not fall within the exception...."

NUREG-1409 also provides an example where an evolving understanding of technical issues resulted in a compliance backfit that was apparently justified for at least some licensees. In

response to industry claims that Bulletin 88-11[71] lacked any backfitting justification, the NRC staff responded:

> Although the justification was not printed in the bulletin, NRC Bulletin 88-11, "Pressurizer Surge Line Thermal Stratification," was justified as a backfit. It is an example of a backfit that was determined by the responsible NRC official to be required as a matter of compliance with existing requirements and commitments. The CRGR reviewed the bulletin and concurred. The regulations currently require licensees to meet the applicable codes of the American Society of Mechanical Engineers (ASME), *Boiler and Pressure Vessel Code*. Because of the NRC staff's concern with the integrity of the surge line, licensees were requested to perform their fatigue analysis in accordance with the latest ASME Section III requirements that incorporate high cycle fatigue analysis. The justification provided by the NRC staff was that previously unconsidered thermal stratification phenomenon may invalidate the existing analysis performed to confirm the integrity of the surge line.
>
> Subsequently, it was understood that some licensees believed that the NRC staff's rationale was in error because they were not committed to the latest ASME Section III requirements by virtue of their license commitment. However, the issue became moot because these licensees undertook the analysis voluntarily in view of the safety importance of the issue and the fact that previous versions of the ASME Code did not completely address the concern.

[71] NRC 1988e

APPENDIX B: QUALIFICATION OF PRESSURE RELIEF VALVES IN NUCLEAR POWER PLANTS IN RESPONSE TO TMI-2 ACCIDENT

Byron and Braidwood Design and Code Requirements

Nuclear power plants in the United States use various types of pressure relief valves to protect personnel and equipment from overpressure events within reactor fluid systems. Pressure relief valves include safety valves, safety relief valves, and relief valves, with different designs, operating conditions, and requirements. The American Society of Mechanical Engineers (ASME) *Boiler and Pressure Vessel Code* (BPV Code), Section III, Division 1, specifies requirements for the design, operation, installation, and testing of pressure relief valves used for various functions in nuclear power plants.[72] For example, the ASME BPV Code (2007 Edition) in Article NB-7000, Overpressure Protection, specifies requirements several service conditions:

- steam and air or gas service for safety valves;
- steam, air or gas, and liquid service for safety relief valves;
- liquid service for relief valves; and
- steam, air or gas, and liquid service for pilot operated or power actuated pressure relief valves.

The ASME *Code for Operation and Maintenance of Nuclear Power Plants* (OM Code) provides requirements for the preservice and inservice testing (IST) programs for pressure relief valves in nuclear power plants.

Braidwood, Units 1 and 2 (Braidwood) and Byron, Units 1 and 2 (Byron) are Westinghouse-designed pressurized-water reactors (PWRs) that received their construction permits under Title 10 of the *Code of Federal Regulations* (10 CFR), Part 50, in December 1975. The pressurizer for each unit is equipped with three pressurizer safety valves (PSVs) and two power-operated relief valves (PORVs). The three PSVs are Crosby Model HP-BP-86, size 6M6 (6-inch), spring-loaded pop type, opened by direct fluid pressure. The PORVs are Copes-Vulcan Model D-100-160 3-inch pneumatic-actuated globe valves that respond to a signal from the pressure sensing system or to manual control. Each PORV can be isolated by a motor-operated block valve.

The ASME BPV Code of record for the design of the PSVs at Byron and Braidwood is the 1971 Edition through the Winter 1972 addenda of the ASME BPV Code, Section III. The ASME BPV Code applicable to Byron and Braidwood includes requirements for overpressure protection, including the following:

- Section NB-7300, "Overpressure Protection Report," in NB-7320(f) requires that the report include the redundancy and independence of the pressure-relief devices and their associated pressure-sensing and controls systems employed to preclude a loss of overpressure protection in the event of a failure of any pressure-relief device, or its sensing element, or its associated control, or an external power source.

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[72] References to individual ASME Code publications are not provided in Appendix D, but they are publicly available from ASME for a fee.

- Paragraph NB-7411, “Relieving Capacity of Pressure-Relief Devices,” specifies that the total rated relieving capacity shall be sufficient to prevent a rise in pressure of more than 10 percent above system design pressure (at design temperature) within the pressure-retaining boundary of the system, under any pressure transient anticipated to arise as summarized in the Overpressure Protection Report.

- Paragraph NB-7421, “Required Number and Capacity of Pressure-Relief Devices for Nuclear Systems,” states that the required relieving capacity intended for overpressure protection of a nuclear power system or portions of the system shall be secured by the use of at least two pressure-relief devices.

At the time of the Byron and Braidwood operating license review, Revision 1 of Standard Review Plan (SRP) Section 15.5.1-15.5.2 and Section 15.6.1 provided general staff guidance for these plant transients.[73] In March 2007, the NRC staff issued Revision 2 to these SRP sections with significantly more detail, including a statement that PSVs and PORVs are assumed to fail open if they relieve water without being qualified.[74]

Actions Following Three Mile Island, Unit 2 Accident

The accident at Three Mile Island, Unit 2 (TMI-2) on March 28, 1979, included failure of a PORV on the pressurizer to reclose properly during the event. Based on lessons learned from the TMI-2 accident, the NRC issued recommendations regarding performance testing of safety and relief valves used in nuclear power plants in NUREG-0578.[75] In particular, the NRC staff recommended in Section 2.1.2, “Performance Testing for BWR [boiling-water reactor] and PWR Relief and Safety Valves,” of NUREG-0578 that nuclear power plant licensees commit to provide performance verification by full-scale prototypical testing for all relief and safety valves.

In October 1980, the NRC issued a letter to all then-operating nuclear power plants and applicants for operating licenses and holders of construction permits forwarding NUREG-0737.[76] TMI Action Plan Item II.D.1 in NUREG-0737 specified the NRC position that PWR and BWR licensees and applicants shall conduct testing to “qualify” the reactor coolant system (RCS) relief and safety valves under expected operating conditions for design-basis transients and accidents. The detailed clarification in NUREG-0737 of this NRC position specified the following:

> Licensees and applicants shall determine the expected valve operating conditions through the use of analyses of accidents and anticipated operational occurrences referenced in Regulatory Guide 1.70, Revision 2. The single failures applied to these analyses shall be chosen so that the dynamic forces on the safety and relief valves are maximized. Test-pressures shall be the highest predicted by conventional safety analysis procedures. [RCS] relief and safety valve qualification shall include qualification of associated control circuitry, piping, and supports, as well as the valves themselves.
>
> A. Performance Testing of Relief and Safety Valves--The following information must be provided in report form by October 1, 1981:

[73] NRC 1981b and NRC 1981c
[74] NRC 2007b and NRC 2007c
[75] NRC 1979a
[76] NRC 1980b and NRC 1980c

(1) Evidence supported by test of safety and relief valve functionability for expected operating and accident (non-[anticipated transient without scram]) conditions must be provided to NRC. The testing should demonstrate that the valves will open and reclose under the expected flow conditions.

(2) Since it is not planned to test all valves on all plants, each licensee must submit to NRC a correlation or other evidence to substantiate that the valves tested in the EPRI (Electric Power Research Institute) or other generic test program demonstrate the functionability of as-installed primary relief and safety valves. This correlation must show that the test conditions used are equivalent to expected operating and accident conditions as prescribed in the final safety analysis report (FSAR). The effect of as-built relief and safety valve discharge piping on valve operability must also be accounted for, if it is different from the generic test loop piping.

(3) Test data including criteria for success and failure of valves tested must be provided for NRC staff review and evaluation. These test data should include data that would permit plant-specific evaluation of discharge piping and supports that are not directly tested.

In describing the type of review to be conducted for this regulatory position, the NRC staff stated the following:

Pre-implementation review will be performed for EPRI and BWR test programs with respect to qualification of relief and safety valves. Also, the applicants' proposal for functional testing or qualification of PWR valves will be reviewed. Post-implementation review will also be performed of the test data and test results as applied to plant-specific situations.

In specifying the documentation required to satisfy this regulatory position, the NRC staff stated the following:

Pre-implementation review will be based on EPRI, BWR, and applicant submittals with regard to the various test programs. These submittals should be made on a timely basis as noted below, to allow for adequate review and to ensure that the following valve qualification dates can be met:

Final PWR (EPRI) Test Program--July 1, 1980

Final BWR Test Program--October 1, 1980

Block Valve Qualification Program--January 1, 1981

Post-implementation review will be based on the applicants' plant-specific submittals for qualification of safety relief valves and block valves. To properly evaluate these plant-specific applications, the test data and results of the various programs will also be required by the following dates:

PWR (EPRI)/BWR Generic Test Program Results--July 1, 1981

Plant-specific submittals confirming adequacy of safety and relief valves based on licensee/applicant preliminary review of generic test program results--July 1, 1981

Plant-specific reports for safety and relief valve qualification--October 1, 1981

Plant-specific submittals for piping and support evaluations--January 1, 1982

Plant-specific submittals for block valve qualification--July 1, 1982

EPRI Testing

In October 1982, EPRI issued NP-2670-LD to address testing of PORVs.[77] This report has been referenced by certain licensees (e.g., Section 15.2.14 of the North Anna, Units 1 and 2 Updated Final Safety Analysis Report (UFSAR)[78]).

In December 1982, EPRI issued NP-2628-SR, which described safety and relief valve tests for types of valves in service at nuclear power plants.[79] In particular, Section 3.5 documented the testing of Crosby safety valves similar to the PSVs at Byron and Braidwood, including two water tests. The report indicated chattering of the safety valves with subsequent inspection finding galled surfaces and damage to internal parts. Section 4.6 documented testing of Copes-Vulcan relief valves similar to the pressurizer PORVs at Byron and Braidwood, although the extent of water testing was not fully described. The report indicated no damage found during the inspection of the Copes-Vulcan relief valves. The report did not indicate any failures of the Crosby or Copes-Vulcan valves to reseat after discharging water during the testing.

EPRI also published NP-2770-LD in the early 1980s to describe the testing of PWR primary system safety valves. Volume 1, issued in December 1982, provides a summary of the test program and its results.[80] Section 4.5 of Volume 1 indicates that the following tests were performed on the Crosby 6M6 PSV: 11 steam tests with filled loop seals, 3 steam-to-water transition tests, and 2 water tests. The report states that the valve experienced chatter during the tests, and one water test had to be terminated. The individual volumes of EPRI NP-2770-LD discuss the test results for each specific PSV type. Volume 6, issued in March 1983, provides the test details for the Crosby 6M6 PSV.

Westinghouse Evaluation of EPRI Testing

In July 1982, the Westinghouse Owners Group (WOG) submitted WCAP-10105.[81] In WCAP-10105, the WOG indicated that the design specification for PSVs in Westinghouse-designed nuclear power plants is for steam service only. Based on a review of the EPRI test data, the WOG concluded that the valves performed with chatter, but did not identify any valve damage.

In January 1988, Westinghouse issued WCAP-11677, which compared the EPRI test data with feedwater line break safety analyses.[82] Westinghouse determined that all nuclear power plants addressed in the EPRI testing had PSVs that would operate reliably during water discharge. Westinghouse evaluated the performance of the Crosby 6M6 PSVs during the EPRI tests, and

[77] EPRI 1982a
[78] VEPCO 2015
[79] EPRI 1982b
[80] EPRI 1982c
[81] WOG 1982
[82] Westinghouse 1988

considered that the performance involved less significant flutter (half lift motion) than the chatter (full lift motion) determined in the EPRI report. Westinghouse concluded that the Crosby 6M6 PSV can pass slightly subcooled water at a minimum up to three times without damage.

Byron and Braidwood Licensing and Response to TMI Requirements

The NRC safety evaluation reports (SERs) associated with the issuance of the operating licenses for Byron and Braidwood included evaluation of the TMI Action Plan items.[83] In the introduction to the Braidwood SER, the NRC staff stated that the review and evaluation of compliance by the applicant with the licensing requirements established in NUREG-0660[84] and TMI Action Plan Item II.D.1 were incorporated into the reviews summarized throughout the SER.

Appendix E, "Requirements Resulting from TMI-2 Accident," to the Byron and Braidwood UFSAR in Section E.23, "Relief and Safety Valve Test Requirements (II.D.1)," references the 1982 transmittal from Consumers Power of a test report for the EPRI safety and relief valve test program.[85] The UFSAR states that the final evaluation of the data indicated that the relief and safety valves will perform their intended functions for all expected fluid inlet conditions. The UFSAR also references the October 1982 licensee evaluation of the adequacy of the relief and safety valves that had been submitted to the NRC.[86]

In Supplement 1 to the Braidwood SER,[87] in Section 3.9.3.3, "Design and Installation of Pressure Relief Devices," the NRC staff stated that EPRI had completed a full-scale valve testing program and referenced the July 1982 submittal of WCAP-10105. The NRC staff stated that the applicant responded to a requirement to demonstrate operability of these valves through submittals dated July 1, 1982, October 26, 1982, and December 30, 1983. On the basis of a preliminary review, the NRC staff concluded that the applicant's general approach to responding to this item was acceptable, and provided adequate assurance that the RCS overpressure protection systems at Braidwood could adequately perform their intended functions. The NRC staff stated that if the detailed review revealed that modifications or adjustments to safety valves, PORVs, PORV block valves, or associated piping, would be needed to ensure that all intended design margins were present, the NRC staff would require that the applicant make appropriate modifications. The NRC staff categorized this issue as a Confirmatory Item. The NRC issued operating licenses for all four Byron and Braidwood Units between February 1985 and May 1988.

Closure of TMI Action Plan Item II.D.1 for Byron and Braidwood

Following the issuance of the operating licenses, the NRC staff documented its review of the response to TMI Action Plan Item II.D.1 for Byron and Braidwood via two letters that transmitted similar Technical Evaluation Reports (TERs) developed by Idaho National Engineering Laboratory (INEL).[88] In its letters, the NRC staff indicated that the licensee should develop and adopt plant procedures to inspect the pressurizer valves after each lift involving loop seal or water discharge. The TERs described the INEL review of the EPRI testing of PSVs and PORVs

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[83] NRC 1983 and NRC 1986b (Braidwood), NRC 1984 and NRC 1987a (Byron)
[84] NRC 1980a
[85] Consumers 1982
[86] ComEd 1982
[87] NRC 1986b. Similar discussion appears in NRC 1984 for Byron, and NRC 1987a (also for Byron) states that TMI Action Plan Item II.D.1 had been closed in NRC 1984.
[88] NRC 1988c (Byron) and NRC 1990a (Braidwood)

similar to the Byron and Braidwood pressurizer valves. The TERs concluded that Byron and Braidwood had provided an acceptable response to TMI Action Plan Item II.D.1.

Section 4.2.3, "Extended High Pressure Injection [HPI] Event," of the TERs stated that the potential for water discharge in extended HPI events can be disregarded for an extended high pressure injection event because at least 20 minutes would be available for operator action.

Water discharge was evaluated, however, in Section 4.2.2, "FSAR Liquid Transients," of the TERs This section discussed the evaluation of the PSVs and PORVs for feedwater line break accidents that would include water discharge, and determined that the EPRI tests were applicable to the Byron and Braidwood PSVs and PORVs.

In addition, Section 4.3.1, "Safety Valves," and Section 4.3.2, "Power Operated Relief Valves," of the TERs determined that the performance of the PSVs and PORVs was acceptable based on the EPRI tests, including water discharge tests. The TERs indicated that the PSV had two applicable tests: a loop seal steam-water transition test where the valve opened, chattered and stabilized to close; and a saturated water test where the valve opened with water, chattered, and stabilized. The TERs indicated that the PORV opened and closed on demand in the loop seal steam-water transition test, with a bending moment that was evaluated by analysis.

APPENDIX C: CONCERNS REGARDING PERFORMANCE OF PRESSURIZER VALVES UNDER WATER FLOW CONDITIONS

Westinghouse Nuclear Safety Advisory Letter

In 1993 and 1994, Westinghouse issued Nuclear Safety Advisory Letter (NSAL) 93-013 its Supplement 1 to operating nuclear power plants (including Byron and Braidwood).[89] These advisories resulted from Westinghouse's discovery that potentially nonconservative assumptions were used in the licensing analysis of the Inadvertent Operation of the Emergency Core Cooling System at Power (IOECCS) event.

In NSAL-93-013, Westinghouse recommended that licensees determine whether their pressurizer safety relief valves (PSRVs) are capable of closing following discharge of subcooled water. Westinghouse noted that the PSRVs might have been designed or "qualified" to relieve subcooled water. Westinghouse indicated that water discharge through the power-operated relief valves (PORVs) is not a concern, because the PORV block valves can be used to isolate the PORVs if they fail to close. If the PSRVs are not designed or qualified for subcooled water discharge, Westinghouse recommended that licensees re-evaluate the IOECCS event with three possible options of (1) reducing emergency core cooling system (ECCS) flow used in the safety analysis, (2) using a less restrictive operator response time, or (3) crediting the use of one or more PORVs to help mitigate the event.

In Supplement 1 to NSAL-93-013, Westinghouse informed licensees of a potential reduced time for operator action if a positive displacement pump is in service. Westinghouse also advised licensees to qualify the PSRVs and the piping downstream of the PSRVs and PORVs if water discharge from the pressurizer were predicted.

Some licensees of operating nuclear power plants informed the NRC of their actions to address the potential concerns regarding water discharge from pressurizer safety valves (PSVs) and PORVs. A sample of actions by nuclear power plant licensees is summarized below in the "Plant-Specific Actions" section.

Additional NRC Generic Communications and Guidance

In 2003, the NRC staff issued a review standard for extended power uprate (EPU) reviews.[90] Item 8 on page 7 of the review standard states that pressurizer level should not be allowed to reach a pressurizer water-solid condition.

In 2005, the NRC issued Regulatory Issue Summary (RIS) 2005-29 to notify nuclear power plant licensees of a concern identified during reviews of power uprate requests.[91] In RIS 2005-29, the NRC staff stated that typically Condition II scenarios[92] involve discharging water through relief or safety valves that are not qualified for water discharge. The NRC staff stated that these valves are then assumed to fail in the open position and create a small-break loss-of-coolant accident (LOCA). The NRC staff stated that it was concerned that some licensees may be crediting PORVs without qualification for water discharge and without establishing additional restrictions to ensure the availability of PORVs and block valves. The NRC staff stated that the

[89] Westinghouse 1993 and Westinghouse 1994
[90] NRC 2003
[91] NRC 2005b
[92] As defined in American Nuclear Society (ANS) Standard 51.1/N18.2-1973 (ANS 1973).

advice in Westinghouse NSAL-93-013 to use the PORV block valves to isolate the PORVs is inconsistent with non-escalation position.

In draft Revision 1 to RIS 2005-29, the NRC staff addresses the specific ANS Condition II scenarios of chemical volume and control system (CVCS) malfunction, inadvertent opening of a PORV or PSV (IOPSRV), and the IOECCS event.[93] Regarding the CVCS malfunction, the NRC staff states that performing only a reactivity anomaly analysis or assuming that this malfunction is not as severe as the IOECCS event is not acceptable. Regarding the IOPSRV event, the NRC staff stated that inadvertent opening of PSV or PORV could continue as an ANS Condition III small break LOCA and fails to meet the non-escalation position. Regarding the IOECCS event, the NRC staff states that five of the alternative approaches in NSAL-93-013 fail to meet the non-escalation position. The NRC staff indicated that these unacceptable alternative approaches are:

1. closing the block valve
2. assuming that the PORV is not operable
3. addressing a stuck-open PORV or PSV as a separate ANS Condition II event
4. determining that a stuck-open PORV or PSV is not as severe as a small break LOCA
5. determining that RCS loss through PORV is made up by ECCS flow

Additional General PSV/PORV Information

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.[94] 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.

In 2011, the NRC summarized relief valve performance data in NUREG/CR-7037, based on a study by the Idaho National Laboratory.[95] With respect to pressurizer PORVs, the report found four separate water discharge events at four PWR plants. The report estimated 698 total demands on these PORVs during their water discharge events with no failures to close. The report also summarized test data for three valve types from the Equipment Performance and Information Exchange (EPIX) database maintained by the Institute of Nuclear Power Operations. The report indicates two failures of PORVs to reclose during 2070 demands, but does not specify water or steam service for the EPIX test information. With respect to PSVs, the report indicates two failures out of four total demands following plant scrams, but does not indicate water or steam service. Following a request by the Panel, NRC staff from the Office of Nuclear Regulatory Research provided Licensee Event Report information indicating that the two PSV failures involved incomplete reseating of the valves with leakage of 25 and 200 gallons per minute, respectively. The report summarized EPIX test data for PSVs as no failures to reclose during 1805 demands.

[93] NRC 2015a
[94] EPRI 2004
[95] NRC 2011

Plant-Specific Actions

Diablo Canyon

In 1996, the licensee for Diablo Canyon Power Plant (Diablo Canyon) submitted a report of its evaluation under Title 10 of the *Code of Federal Regulations* (10 CFR), Section 50.59, "Changes, tests and experiments," of the potential for an IOECCS event.[96] The submittal included NSAL-93-013 and its Supplement 1 as enclosures. The licensee indicated that the PSVs had not been initially qualified for water discharge, but were subsequently qualified to discharge water for a brief period. The licensee indicated that WCAP-11677 was applicable and demonstrated that the PSVs were operable.

In 2004, the NRC issued a license amendment for Diablo Canyon that allowed credit for actuation of the PORVs in response to inadvertent safety injection (SI) actuation, to avoid challenges to the PSVs.[97] To support the NRC staff's review, the licensee submitted additional information related to the capability of the PORVs to function adequately under conditions predicted for design-basis transients and accidents.[98] In response to a question regarding the design adequacy of the PORVs if the pressurizer becomes water solid, the licensee referenced a January 1986 NRC letter that had accepted the adequacy of the PORV and block valve design and confirmatory testing for a range of fluid conditions (full pressure steam, steam to water transition, and subcooled water fluid).[99]

Salem

In 1997, the NRC issued a license amendment revising the technical specification (TS) for Salem Nuclear Generating Station, Units 1 and 2 (Salem) to ensure that the automatic capability of the PORVs to relieve pressure would be maintained.[100] In response to NSAL-93-013, the licensee determined that an inadvertent SI actuation at power could cause the pressurizer to become water solid. The PSVs would lift and discharge water if the automatic operation of the PORVs were not made available for reactor coolant system (RCS) depressurization early in the transient. In that the Salem PSVs were not designed to relieve water, it was noted that water discharge could cause the PSVs to fail in the open position.

During the review, the NRC staff noted that the PORVs were not designed to "safety related" standards and, thus, could not be credited for automatic mitigation of an inadvertent SI actuation at power. In response, the licensee proposed an upgrade of the PORVs to eliminate the possibility that a single active failure of a PORV component could prevent the mitigation of an inadvertent SI actuation at power. As discussed in the NRC staff's safety evaluation (SE), the licensee implemented modifications to the PORV circuitry to qualify the upgraded circuitry as safety-related.

Regarding PORV performance, the licensee evaluated the PORV air accumulators and determined that they had sufficient capacity for the inadvertent SI event. The licensee also reported that endurance tests had been performed with five different trims (with different trim materials) on one PORV at Wyle Laboratories to demonstrate that (1) after 2000 consecutive

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[96] PG&E 1996
[97] NRC 2004a
[98] PG&E 2003
[99] NRC 1986a
[100] NRC 1997

operations, there were no packing leaks or packing gland adjustments required; (2) there was no diaphragm failure; and (3) the solenoid valve withstood 10,000 operations without any loss of function. Based on this information, the NRC staff concluded that the PORV performance was acceptable to mitigate an inadvertent SI event.

Millstone 3

In 1998, the NRC issued a license amendment for Millstone Nuclear Power Station, Unit 3 (Millstone 3) that revised the TS to ensure that the capability of the PORVs to relieve pressure would be maintained.[101] The revised TS Bases stated that the PORVs and their associated piping had been demonstrated to be "qualified" for water discharge. The PORVs would prevent water discharge from the PSVs, for which qualification for water discharge had not been demonstrated. The TS Bases also stated that the prime importance for the capability to close the block valve is to isolate a stuck-open PORV. In the SE, the NRC staff referenced a December 1997 Licensee Event Report that notified the NRC of the issue of potential failure of PSVs following water discharge.[102]

As part of this license amendment, the licensee upgraded the PORV circuitry, added additional PORV surveillance requirements, qualified the PORVs and associated piping for water discharge, and revised emergency procedures to allow plant operators additional time to terminate the event. With respect to the PORV circuitry, the NRC staff concluded that the PORV circuitry modifications qualified the PORV control circuitry as safety-related. With respect to PORV performance, the licensee reanalyzed the inadvertent SI event with the LOFTRAN computer code to determine the time available for operator action to make a PORV available and provide the mass and energy releases needed to qualify the PORVs and associated piping for water discharge. The licensee referenced EPRI testing that was said to generically resolve TMI Action Plan Items associated with PORVs and safety valve qualification for water and steam discharge, specifically the results from four tests of a Garrett PORV (such as used at Millstone 3) for water discharge.[103] The licensee determined that the PORVs and associated piping are qualified for 1 hour of water discharge for an IOECCS event. The licensee also stated that the PORV manufacturer performed numerous cycle tests to verify the performance of the valve design, and also verified that valve seat leakage was acceptable. The licensee stated that the PORV block valves had been evaluated for water discharge in accordance with the program established in response to Generic Letter (GL) 89-10.[104] The NRC staff found the licensee information regarding the qualification of the PORVs for water discharge during the inadvertent SI event to be acceptable.

Callaway

In 2000, the NRC issued a license amendment for Callaway Plant, Unit 1 (Callaway) that revised the TS to change the PSV lift setting range.[105] The changes also credited automatic actuation of at least one PORV during an IOECCS event to prevent water discharge through the PSVs; to enable this credit, the licensee modified and upgraded the PORV circuitry to full Class 1E. In its license amendment request,[106] the licensee had stated that the design function

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[101] NRC 1998
[102] Northeast 1997
[103] EPRI 1982a (Volume 11)
[104] NRC 1989
[105] NRC 2000
[106] Union Electric 2000

of the valves was not being changed and the conclusions documented in the NRC staff's previous evaluation of Callaway's response to TMI Action Plan Item II.D.1[107] were also unchanged. As a result, the licensee stated that the PORVs and associated discharge piping can accommodate water discharge.

Byron and Braidwood

In 1998, the licensee for Byron and Braidwood requested an amendment to its TS to take credit for automatic operation of the PORVs to mitigate an IOECCS event.[108] In the amendment request, the licensee stated that the PSVs had not been qualified to reseat after passing subcooled liquid. The licensee stated that the PORVs at Byron and Braidwood are safety-related components with safety-related actuators and accumulator tanks, with PORV control circuits classified as safety-related. The licensee noted that some portions of the PORV circuitry are nonsafety-related, with improvements implemented in response to GL 90-06.[109] The licensee stated that the PORV block valves are within the scope of the GL 89-10 program.

In 1999, the NRC staff requested additional information related to concerns that the PORV circuitry did not meet the single failure criterion.[110] The licensee reevaluated its approach and withdrew its TS amendment request.[111] No further action regarding this amendment request was identified by the Panel. However, in a public meeting during the review of the NRR Appeal,[112] the licensee stated that the PORVs and their block valves at Byron and Braidwood are safety-related with the exception of one circuitry aspect of the PORV.[113]

In 2001, the NRC issued a license amendment for Byron and Braidwood to increase the maximum thermal power for each unit from 3411 megawatts thermal (MWt) to 3586.6 MWt (commonly referred as a stretch power uprate).[114] During its review, the NRC staff requested that the licensee address water solid conditions in the pressurizer, because it had generally not accepted a solid pressurizer for an IOECCS event to order given the potential for all three PSVs to be stuck open due to liquid relief through these safety valves. In response, the licensee stated that Section 15.5.1, "Inadvertent Operation of Emergency Core Cooling System During Power Operation," of the UFSAR had been revised to credit the PSVs to pass water.[115] The licensee discussed the EPRI testing program in response TMI Action Plan Item II.D.1, with the results summarized in EPRI NP-2628-SR.[116] The licensee referenced previous NRC approvals related to TMI Action Plan Item II.D.1.[117]

The NRC staff made a further request regarding the temperature of water that would be discharged by the PSVs and the length of time that the PSVs would be expected to discharge water. The NRC staff also asked the licensee to discuss which EPRI tests are applicable to the Byron and Braidwood condition. In response, the licensee stated that the PSVs would close after discharging water, although they may not be leaktight.[118] The licensee stated that the

[107] NRC 1987b
[108] ComEd 1998
[109] NRC 1990b
[110] NRC 1999
[111] ComEd 1999
[112] Exelon 2015
[113] NRC 2016a
[114] NRC 2001b
[115] ComEd 2000b
[116] EPRI 1982b
[117] NRC 1998c and NRC 1990a

leakage from up to three leaking PSVs is bounded by one fully open PSV. The licensee indicated that the EPRI testing of the Crosby safety valves in EPRI NP-2770-LD, Volumes 1 and 6,[119] are applicable. The licensee indicated that valve chatter occurred during the tests with damage to the internals, but that the safety valve closed in response to system depressurization. The licensee stated that the Byron and Braidwood pressurizer water temperature of 590 °F is higher than the EPRI tests (530 °F). The licensee stated that the assumed length of the event is 20 minutes from initial SI signal to when the system pressure is restored below PSV lift setpoint.

In Section 3.2 of the SE accompanying the license amendment, the NRC staff discussed its review of the performance of the PORVs and PSVs to discharge liquid water for approximately 20 minutes. The NRC staff discussed the EPRI testing program, with the conclusion that the PSV would close in response to system depressurization. The NRC staff reviewed the licensee's evaluation of the performance of the PSVs for liquid water conditions. The NRC staff found that the EPRI tests adequately demonstrated the performance of the valves for the expected water temperature conditions, and that there was reasonable assurance that the valves would adequately reseat following the spurious SI event. The NRC staff determined that EPRI test data indicated that the PSVs might chatter for the expected fluid inlet temperature, but that the resulting PSV seat leakage following the water discharge would be less than the discharge from one stuck-open PSV. Therefore, the NRC staff found the licensee's crediting of the PSVs to discharge liquid water during the spurious SI event to be acceptable. This portion of the SE was based on input provided by the Office of Nuclear Reactor Regulation (NRR) Reactor Systems Branch, with technical input from the responsible staff member for safety valves in the NRR Division of Engineering.[120]

As noted by the licensee, Section 15.5.1 of the Byron and Braidwood UFSAR at the time of the stretch power uprate includes PSV water discharge and references the TMI Action Plan Item II.D.1 approvals.[121] The current UFSAR Revision 15 concludes that the IOECCS event does not progress into a stuck-open PSV LOCA event.[122] The UFSAR states that all three PSVs may lift but will reclose, and that the leakage is bounded by one fully open valve with the consequences bounded by the IOPSRV event. The UFSAR also specifies that if SI results in discharge of coolant through the pressurizer valves, the operators will bring the plant to cold shutdown to inspect the valves.

In 2004, the NRC issued a license amendment for Byron and Braidwood granting an adjustment to the PSV setpoints.[123] As documented in the SE, the NRC staff requested during its review that the licensee perform a quantitative analysis regarding PSV water cycles and discharge water temperature. For the loss of ac power (LOAC) with reactor coolant pump (RCP) seal injection event, the licensee's analysis indicated that continued injection of water into the RCS through the RCP seals would result in a water-solid pressurizer and water discharge through the PSVs. The proposed PSV setpoint tolerance assuming negative tolerance would result in a lower PSV lift setpoint. With the lower setpoint, the PSV would open earlier, and a larger number of PSV water cycles with a lower water discharge temperature could result during the transient. The licensee performed an analysis of the LOAC with RCP seal injection event, and

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[118] Exelon 2001
[119] EPRI 1982c and EPRI 1983
[120] NRC 2001a
[121] Exelon 2002
[122] Exelon 2014
[123] NRC 2004b

determined the revised PSV setpoint would result in an increase of about one PSV water cycle and a reduction in the water discharge temperature of about 0.5 °F. A comparison of the reanalysis showed that the spurious SI event remained the limiting event since it resulted in a greater increase in the number of PSV water cycles (two cycles vs. one cycle) and a greater decrease in the PSV discharge water temperature (3.0 °F vs. 0.5 °F) than that calculated for the LOAC with RCP seal injection event. The water discharge temperature in the analysis of record for the spurious SI event was 590 °F. The lowest discharge water temperature for the spurious SI event with the revised PSV setpoint was 587 °F. The NRC staff found that the calculated water discharge temperature (587 °F) was significantly higher than the discharge water temperature of 530 °F that was used to support operability of the PSVs as discussed in the analysis of record. As a result, the NRC staff concluded that the analysis was acceptable to assure that the PSVs will remain operable following a spurious SI event.

In 2014, the NRC issued a license amendment for Byron and Braidwood granting a measurement uncertainty recapture (MUR) power uprate.[124] The NRC staff determined that the IOECCS event was outside of the scope of the MUR power uprate, because the licensee did not propose to modify the Chapter 15 analyses related to PSV and PORV water discharge.

With respect to inservice testing (IST) activities, the Byron IST program[125] references the ASME *Code for Operation and Maintenance of Nuclear Power Plants* (OM Code), 2004 Edition through 2006 Addenda; and the Braidwood IST program[126] references the ASME OM Code, 2001 Edition through 2003 Addenda. The Byron IST Program specifies the following testing and intervals for the PORVs, PORV block valves, and PSVs:

- **PORV:** fail safe test closed (cold shutdown interval), stroke-time exercise open and closed (cold shutdown interval), and position indication test (2 year interval)

- **PORV Block Valve:** exercise open and closed (2 year interval); position indication test (Joint Owners Group (JOG) Program interval); and open and closed test in accordance with ASME OM Code Case OMN-1, "Alternative Rules for Preservice and Inservice Testing of Active Electric Motor Operated Valve Assemblies in Light-Water Reactor Power Plants" (JOG Program interval)

- **PSV:** position indication test (2 year interval) and relief valve test (5 year interval), referencing ASME OM Code, Appendix I, "Inservice Testing of Pressure Relief Devices in Light-Water Reactor Nuclear Power Plants"

The Braidwood IST Program specifies the following testing and intervals for the PORVs, PORV block valves, and PSVs:

- **PORV:** fail safe test closed (refueling outage interval), stroke-time exercise open and closed (refueling outage interval), and position indication test (2 year interval).

- **PORV Block Valve:** exercise open and closed (quarterly interval) and position indication test (2 year interval)

[124] NRC 2014a
[125] Exelon 2016b
[126] Exelon 2009

- **PSV:** position indication test (2 year interval), and relief valve test (5 year interval), referencing ASME OM Code, Appendix I

Shearon Harris

In 2001, the NRC issued a license amendment to Shearon Harris Nuclear Power Plant, Unit 1 (Shearon Harris) for steam generator replacement and a power uprate to a maximum power level of 2900 MWt (approximately 4.5 percent).[127] In addressing the licensee's evaluation of Standard Review Plan (SRP) Section 15.5.1, the NRC staff found that the analysis showed that the calculated inlet pressures and temperatures required for the PORVs and safety relief valves (SRVs)[128] to operate in a water environment were within the valve operable ranges, and thus ensured that the PORV and SRV would be operable during the transient. The valve operable ranges were previously determined by the licensee to support operability of the PORV and SRV during the discharge of subcooled water in accordance with the TMI Action Plan Item II.D.1 requirements. Based on the analysis meeting the acceptance criteria of SRP Section 15.5.1 with respect to the RCS pressure limit and departure-from-nucleate-boiling limit, the NRC staff concluded that the analysis was acceptable.

Beaver Valley

In 2006, the NRC issued a license amendment authorizing an EPU for Beaver Valley Power Station, Units 1 and 2 (Beaver Valley), an approximate 8-percent increase in thermal power to 2,900 MWt.[129] In the SE accompanying the amendment, the NRC staff described its review of the capability of the PSVs to discharge liquid and adequately reseat for a spurious SI actuation. The NRC staff specifically evaluated whether the PSVs could reasonably be expected to reseat to prevent the spurious SI actuation (an ANS Condition II event) from causing a stuck-open PSV (an ANS Condition III event). This issue was said to be further discussed in RIS 2005-29. While the PSVs for Beaver Valley were qualified to discharge steam, if the valves discharged water with sufficient subcooling, the NRC staff was concerned that they might not reseat properly.

Based the licensee's analysis, during a spurious SI event, the PSVs would be required to discharge steam followed by high temperature water after the pressurizer filled. The licensee provided plots of the pressurizer water temperatures for this event that indicated that the minimum temperature of the discharged liquid for Beaver Valley was approximately 620 °F. To evaluate the capability of the valves to discharge and reseat, the NRC staff reviewed the available data from the full-flow tests performed during the EPRI test program in 1981 for the specific PSV models representative of those installed at Beaver Valley. The licensee also used the methodology contained in WCAP-11677 and determined that the minimum acceptable liquid temperature for which the PSVs were expected to successfully discharge and reseat was less than the minimum expected temperature for the spurious SI event for Beaver Valley.

The NRC staff agreed that both the minimum expected water discharge temperature and the minimum acceptable water temperature had been conservatively calculated. Therefore, the NRC staff determined that, for purposes of preventing the occurrence of a more serious ANS Condition III event, there was reasonable assurance that the PSVs would discharge water and reseat adequately following a spurious SI actuation. A consideration of the NRC staff in making

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[127] NRC 2001d
[128] This term is used in the Shearon Harris SE; the Panel considers the term SRV to be equivalent to PSV for this facility.
[129] NRC 2006

this finding was that, in the unlikely event of a stuck-open PSV, the ECCS was fully capable of mitigating the resulting LOCA.

Turkey Point

In 2012, the NRC issued a license amendment authorizing an EPU for Turkey Point Nuclear Generating, Units 3 and 4 (Turkey Point), increasing the thermal power level of each unit approximately 15 percent to 2644 MWt.[130]

In the SE accompanying the amendment, the NRC staff indicated that ECCS actuation was not a possible initiator of inadvertent increase in reactor coolant inventory because the high head SI pumps have a shut-off head below the normal RCS operating pressure. The NRC staff stated that a CVCS malfunction that increases RCS inventory was evaluated for the effects of adding water inventory to the RCS. If the pressurizer filled and caused water to be relieved through the PORVs or PSVs, then these valves could stick open and create a small break LOCA. The NRC staff stated that this would violate the acceptance criterion that prohibits the escalation of an anticipated operational occurrence (AOO) into a more serious event. Satisfaction of this acceptance criterion was demonstrated by showing that sufficient time would exist for the operator to recognize the situation and end the charging flow before the pressurizer could fill. The NRC staff concluded that the licensee's analyses of IOECCS and CVCS events adequately accounted for operation of the plant at the proposed power level.

Regarding an inadvertent opening of a PORV, the licensee initially proposed that the consequences of this event were bounded by the small break LOCA. The NRC staff did not accept this proposed disposition. If action were not taken to secure the open valve by either closing the PORV or its block valve, the NRC staff stated that this event could escalate to a small-break LOCA, which would be contrary to the non-escalation position. When the pressurizer filled, water would begin to flow through the open PORV. If the PORV were not qualified for water discharge, the NRC staff stated that it was likely the PORV would not close upon demand. In this way, the NRC staff stated that the inadvertent opening of a PORV, an AOO, would become a small break-LOCA at the top of the pressurizer, an ANS Condition III event. The NRC staff requested that the licensee address the inadvertent opening of the PORV with respect to the third criterion for an ANS Condition II event.

The licensee provided an analysis performed largely in accordance with NRC-approved, Westinghouse analytic methodology using the RETRAN computer code; however, this analysis was performed assuming that the PORV opened instead of the PSV. The NRC staff stated that assuming the opening of the PORV is acceptable, because the PSV is differently qualified, and reseats mechanically. An additional independent fault would be required to cause the PSV to fail to close. The analysis indicated that the pressurizer would fill within about 240 seconds. The licensee stated that there were multiple alarms to indicate the opening of a PORV. The licensee stated that a prompt operator action would be needed to close the PORV and, if the PORV does not close, the operator would be directed to close the block valve. Because the necessary actions would be prompt and simple, the NRC staff agreed that there would be sufficient time to secure the inadvertently open PORV without filling the pressurizer.

[130] NRC 2012a

St. Lucie

In 2012, the NRC issued a license amendment authorizing an EPU for St. Lucie Plant, Unit 2 (St. Lucie, Unit 2) that increased the authorized thermal power level about 12 percent to 3020 MWt.

Regarding an IOECCS event, the high pressure SI pumps would be incapable during power operations of delivering flow to the RCS because the pumps' shut-off head would be less than the normal RCS operating pressure of 2250 pounds per square inch absolute. Therefore, the licensee determined that the inadvertent operation of the ECCS at power event was not a credible event and did not analyze it for the proposed EPU. The NRC staff found that the licensee's position for not analyzing the IOECCS event to be acceptable.

Regarding a CVCS malfunction, the licensee evaluated it as an AOO for the effects of adding water inventory to the RCS. The NRC staff reviewed the licensee's analyses of the CVCS malfunction event and concluded that the licensee's analyses adequately accounted for operation of the plant at the proposed power level and were performed using acceptable analytical models. The NRC staff determined that the licensee's analysis demonstrated that the pressurizer did not become water solid, assuring no water was discharged through the PSVs.

Regarding an IOPORV event, the NRC staff stated that, when viewed from the mass addition perspective, this event could be evaluated in two phases: (1) an inadvertent opening of a pressurizer relief valve, followed by (2) an inadvertent ECCS actuation. In the first phase, the NRC staff stated that this event could be mitigated by closing the open PORV or its block valve. If the PORV or its block valve was not closed, the NRC staff stated that the IOPORV event would enter the second phase with actuation of the ECCS. Based on its review, the NRC staff determined that the pressurizer overfill analysis, available alarming system, and procedures, in combination with simulator exercise results, provided reasonable assurance that the pressurizer would not be expected to fill to a water solid condition that could prevent the PORV or PSV from closing after they were open. The NRC staff therefore concluded that the event would not generate a more serious plant condition, meeting the non-escalation criterion. The NRC staff stated that it reviewed the licensee's analyses of the inadvertent opening of a pressurizer PORV event, and concluded that the licensee's analyses adequately accounted for operation of the plant at the proposed power level and were performed using acceptable analytical models.

The NRC staff concluded that the licensee demonstrated that all AOO acceptance criteria were satisfactorily met.

North Anna

North Anna Power Station, Units 1 and 2 (North Anna) UFSAR Section 15.2.14, "Spurious Operation of the Safety Injection System at Power," describes plant response to an inadvertent SI event. In particular, UFSAR Section 15.2.14.2.3, "Event Propagation," states the following:

> Safety valve (Reference 18) and PORV (Reference 19) testing has revealed no instances of failure of the valves to reseat following water relief. Resulting leakage is within the capacity of the normal makeup system and is therefore not considered to be a small break loss of reactor coolant event. Therefore, the complete filling of the pressurizer and/or water relief via a safety valve as a result of a spurious safety injection does not constitute a failure to meet the event propagation acceptance criterion. Although primary credit for preventing the propagation of the event to a small break loss of reactor coolant event is the

reseating of the PORVs and safety valves, it is noted that the PORVs (which open prior to the safety valves and, if open, preclude safety valve actuation for this event) are provided with block valves which the operator will close in the event of excessive PORV leakage.

North Anna UFSAR Section 15.2.14.3, "Conclusions," indicates that the complete filling of the pressurizer and water discharge via a PSV as a result of a spurious SI does not constitute a failure to meet the non-escalation position. Furthermore, UFSAR Section 15.2, "References," lists EPRI NP-2770-LD and EPRI NP-2670-LD.

Conclusion

In conclusion, the reliance by the licensee for Byron and Braidwood on the acceptable performance of the PSVs and PORVs following water discharge in response to abnormal events is not inconsistent with similar approaches by some other nuclear power plant licensees. In general, the review of activities by various nuclear power plant licensees related to PSV and PORV performance revealed reliance on EPRI, Wyle, and valve vendor testing to provide support for the performance of these valves under various service conditions. Specific certification for flow capacity of these valves for water discharge in accordance with the ASME BPV Code and National Board was not identified in the review of various justifications prepared by nuclear power plant licensees.

In evaluating the historical documents for Byron and Braidwood, the Panel found it challenging to determine specifically how the licensee resolved the concern raised in NSAL-93-013 in its analyses and plant operations. While the record does support a compliance backfit in this case, if (as recommended by the Panel) the NRC staff undertakes a generic review of licensees' treatment of the potential for pressurizer valve damage following water discharge, it may be appropriate to consider what actions have been taken, how operating experience with water discharge has been considered, and how analysis assumptions are considered in operational practices (including inservice testing) at each plant.

APPENDIX D: REFERENCES

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85. **NRC 2016c:** U.S. NRC, memorandum from Marissa G. Bailey to William M. Dean, "Backfit Review Panel Recommendation Regarding Exelon Appeal of Backfit Affecting Byron and Braidwood Stations Regarding Compliance with 10 CFR 50.34(b), GDC 15, GDC 21, GDC 29, and the Licensing Basis," dated March 25, 2016. ADAMS Accession No. ML16082A542 [non-public].

86. **NRC 2016d:** U.S. NRC, letter from William M. Dean to J. Bradley Fewell, Exelon Generation Company, LLC, "U.S. Nuclear Regulatory Commission Response to Backfit Appeal - Braidwood Station, Units 1 and 2, and Byron Station, Units 1 and 2," dated May 3, 2016. ADAMS Accession No. ML16095A204.

87. **NRC 2016e:** U.S. NRC, memorandum from Victor M. McCree to Gary M. Holahan, K. Steven West, Thomas G. Scarbrough, and Michael A. Spencer, "Charter for Backfit Appeal Review Panel Associated with Byron and Braidwood Compliance with 10 CFR 50.34(b), GDC 15, GDC 21, GDC 29, and the Licensing Basis," dated June 22, 2016. ADAMS Accession No. ML16173A311 [non-public].

88. **NRC 2016f:** U.S. NRC, "An Assessment of Core Damage Frequency for Byron/Braidwood Nuclear Power Plants Supporting Backfit Appeal Review Panel," dated August 11, 2016. ADAMS Accession No. ML16214A199 [non-public].

89. **PG&E 1996:** PG&E, letter from Gregory M. Rueger to U.S. NRC, "Forwards completed Licensing Basis Impact Evaluation of FSAR Update change which contains SE performed IAW 10CFR50.59 re reanalysis of inadvertent ECCS actuation accident," dated August 13, 1996. [This document is not in ADAMS, but is available through the NRC Public Document Room using Accession No. 9608200112 and microfiche location 89419:294-322.]

90. **PG&E 2003:** Pacific Gas and Electric Company, letter from David H. Oatley to U.S. NRC, "Response to NRC Request for Additional Information Regarding License Amendment Request 01-018, 'Credit for Automatic Actuation of Pressurizer Power Operated Relief Valves; Pressurizer Safety Valve Loop Seal Temperature,'" dated November 21, 2003. ADAMS Accession No. ML033360735.

91. **TVA 1983:** Tennessee Valley Authority (TVA), letter from L.M. Mills to E. Adensam, U.S. NRC, enclosing "Watts Bar Nuclear Plant Units 1 and 2 Safety Valve Sizing," dated April 18, 1983. ADAMS Accession No. ML080360226.

92. **Union Electric 2000:** Union Electric Company, letter from Alan C. Passwater to U.S. NRC, "Revision to Technical Specifications 3.3.2, 3.4.10, and 3.4.11 – Pressurizer Safety Valves and PORVs," dated May 25, 2000. ADAMS Accession No. ML003719636.

93. **VEPCO 2009:** Virginia Electric and Power Company (VEPCO), letter from L.N. Hartz to U.S. NRC, "North Anna Power Station Units 1 and 2 Updated Final Safety Analysis Report Revision 45," dated October 1, 2009. ADAMS Accession No. ML092810047.

94. **VEPCO 2015:** Virginia Electric and Power Company, letter from Gianna C. Clark to U.S. NRC, "North Anna Power Station Units 1 and 2 Updated Final Safety Analysis Report Revision 51," dated September 30, 2015. ADAMS Accession No. ML15296A098 [non-public].

95. **Westinghouse 1988:** Westinghouse, R.J. Dickinson and J.G. Bass, "Pressurizer Safety Relief Valve Operation for Water Discharge during a Feedwater Line Break," WCAP-11677, dated January 1988. (Submitted to the NRC as part of a May 8, 1989, response to a request for additional information related to Seabrook.) [This document is not in ADAMS, but is available through the NRC Public Document Room using Accession No. 8905120191 and microfiche location 49755:336 – 49756:017.]

96. **Westinghouse 1993:** Westinghouse, Nuclear Safety Advisory Letter (NSAL) 93-013, "Inadvertent ECCS Actuation at Power," dated June 30, 1993. [This document is not in public ADAMS, but is available through the NRC Public Document Room using Accession No. 9608200112 and microfiche location 89419:311-315, as well as in non-public ADAMS Accession No. ML052930330, pages 2-6 of file.]

97. **Westinghouse 1994:** Westinghouse, NSAL-93-013, Supplement 1, "Inadvertent ECCS Actuation at Power," dated October 28, 1994. ADAMS Accession No. ML050320117 [pages 9-15 of file].

98. **Westinghouse 2000:** Westinghouse, NSAL-00-013, "CVCS Modeling Assumption for Loss of Offsite Power Analyses," dated August 23, 2000. ADAMS Accession No. ML103200150 [pages 131-139 of file].

99. **Westinghouse 2007:** Westinghouse, NSAL-07-10, "Loss-of-Normal Feedwater Loss-of-Offsite AC Power Analysis PORV Modeling Assumptions," dated November 7, 2007. ADAMS Accession No. ML100140163 [pages 23-26 of file].

100. **Westinghouse 2011:** Westinghouse, "AP1000 Design Control Document," Revision 19, dated June 13, 2011. ADAMS Accession No. ML11171A500.

101. **WOG 1982:** Westinghouse Owners Group (WOG), letter from Oliver D. Kingsley, Alabama Power Company, to Harold R. Denton, U.S. NRC, "NUREG-0737, Item II.D.1, 'Pressurizer Safety Valve Operability,'" dated July 27, 1982. Forwards Westinghouse WCAP-10105, "Review of Pressurizer Safety Valve Performance as Observed in the EPRI Safety and Relief Valve Test Program," dated June 1982. [This document is not in ADAMS, but is available through the NRC Public Document Room using Accession No. 8208190307 and microfiche location 14387:189-301.]

Report of the Backfit Appeal Review Panel Chartered by the Executive Director for Operations to Evaluate the June 2016 Exelon Backfit Appeal

August XX, 2016

ADAMS Accession No. MLXXXXXXXXX

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1 BACKGROUND

On June 22, 2016,[1] in accordance with NRC Management Directive (MD) 8.4,[2] the NRC Executive Director for Operations (EDO) established a Backfit Appeal Review Panel (Panel) to review the appeal by Exelon Generation Company, LLC (Exelon or the licensee) of the U.S. Nuclear Regulatory Commission (NRC) staff's determination that a backfit is necessary at Braidwood Station, Units 1 and 2 (Braidwood) and Byron Station, Units 1 and 2 (Byron), as well as the NRC staff's application of the compliance backfit exception provided in Title 10 of the *Code of Federal Regulations* (10 CFR), Section 50.109, "Backfitting."

This backfit determination is documented in an October 9, 2015, letter (referred to as the Backfit Letter).[3] The letter describes the NRC staff's review of licensing basis documents for Byron and Braidwood. The NRC staff determined that Byron and Braidwood were not in compliance with the plant-specific design bases and several NRC regulations:

- General Design Criterion (GDC) 15, "Reactor coolant system design," in 10 CFR Appendix A, "General Design Criteria for Nuclear Power Plants"
- GDC 21, "Protection system reliability and testability"
- GDC 29, "Protection against anticipated operational occurrences"
- Paragraph (b) of 10 CFR 50.34, "Contents of applications; technical information"

Specifically, the NRC staff determined that Byron and Braidwood do not comply with provisions in American Nuclear Society (ANS) Standard 51.1/N18.2-1973[4] for ensuring that ANS Condition II events[5] do not progress to more serious ANS Condition III events following water discharge[6] through certain valves. The NRC staff acknowledged that the NRC staff position differed from a previous staff position documented in a May 4, 2001, safety evaluation (SE) supporting a stretch power uprate (referred to as the Uprate SE).[7] However, the NRC staff determined that the backfitting was justified under the compliance exception in 10 CFR 50.109(a)(4)(i). The licensee was directed to take action to resolve the non-compliance.

On December 8, 2015, the licensee appealed the NRC staff's decision to the Director of the Office of Nuclear Reactor Regulation (NRR), stating its disagreement with the NRC's conclusion that the compliance exception to the backfit rule applies in this case, and that the NRC has twice approved the underlying analysis.[8] The referenced approvals were an August 26, 2004, license amendment associated with pressurizer safety valve (PSV) setpoints[9] and the above-

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[1] NRC 2016e (Author and year citations in footnotes refer to the designation of references in Appendix D to this report.)
[2] NRC 2013
[3] NRC 2015b – referred to as the Backfit Letter in the remainder of the report
[4] ANS 1973
[5] Specifically, inadvertent operation of the emergency core cooling system (IOECCS), malfunction of the chemical and volume control system (CVCS), and inadvertent opening of a pressurizer safety or relief valve.
[6] For consistency in this report, the Panel uses the phrase "water discharge" rather than "water relief" or "liquid discharge" (except in direct quotes), as this is the phrase used in the Westinghouse documents that raised the issue addressed in this report.
[7] NRC 2001b – referred to as the Uprate SE in the remainder of the report
[8] Exelon 2015 – referred to as the NRR Appeal in the remainder of the report

referenced Uprate SE. In a letter dated May 3, 2016, the NRC responded to the licensee's appeal and reaffirmed its decision that the backfit per the compliance exception provisions of 10 CFR 50.109(a)(4)(i) is appropriate.[10]

On June 2, 2016, the licensee again appealed the NRC staff's decision, this time to the EDO.[11] The purpose of this report by the Backfit Appeal Review Panel is to provide information and recommendations to support the decision of the EDO.

1.1 Conduct of the Panel's Review

In order to establish a technically sound, well informed, and legally defensible basis for its recommendations, the Backfit Appeal Review Panel undertook a review of the relevant documents in this case. This included the licensee and NRC staff letters mentioned above; the Uprate SE and Setpoint SE; and a June 16, 2016, letter from the Nuclear Energy Institute (NEI)[12] supporting the Exelon backfit appeal. The Panel also reviewed many other related documents, which fall into five broad categories:

- The Backfit Rule (10 CFR 50.109), related court actions, and Commission and staff guidance on application of the Backfit Rule
- Docketed communications for Byron and Braidwood from 1982 to the present, including license amendment requests (LARs) by the licensee, NRC-issued license amendments, NRC requests for additional information (RAIs), licensee responses, meeting summaries, NRC SEs, and the licensee's Updated Final Safety Analysis Report (UFSAR)
- NRC guidance relevant to the analysis of IOECCS events over the period of 1981 to the present, including Standard Review Plan (SRP) Section 15.0, Section 15.5.1 – 15.5.2, and Section 15.6.1[13]
- Westinghouse Nuclear Safety Advisory Letter (NSAL) 93-013[14] and its Supplement 1[15], as well as docketed communications regarding actions taken by other licensees in response to Westinghouse NSAL-93-013
- The history of NRC and industry activities related to power operated relief valves (PORVs), their block valves, and PSVs (including Three Mile Island (TMI) Action Plan Items II.D.1, II.D.3,II.G.1, II.K.3 documented in NUREG-0737[16], as well as Generic ~~Letter~~ Letter 89-10[17] and its supplements), Electric Power Research Institute (EPRI) valve testing, and operating experience (NUREG/CR-7037[18])

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[9] NRC 2004b – referred to as the Setpoint SE in the remainder of the report
[10] NRC 2016d – referred to as NRR Appeal Decision in the remainder of the report
[11] Exelon 2016a – referred to as EDO Appeal in the remainder of the report
[12] NEI 2016
[13] NRC 1981a, NRC 1981b, NRC 1981c, NRC 2007a, NRC 2007b, and NRC 2007c
[14] Westinghouse 1993
[15] Westinghouse 1994
[16] NRC 1980c~~b~~ – referred to as the TMI Action Plan in the remainder of the report; lessons learned from TMI were also presented in NUREG-0578 (NRC 1979a), NUREG-0585 (NRC 1979b), and NUREG-0660 (NRC 1980a)
[17] NRC 1989
[18] NRC 2011

In addition to the document review, the Panel had the benefit of meetings with NRR (both the Division of Safety Systems and the Division of Engineering), the Office of the General Counsel, and the NRC Committee to Review Generic Requirements (CRGR). Both Exelon (Bradley Fewell, Senior Vice President of Regulatory Affairs) and NEI (Tony Pietrangelo, Senior Vice President and Chief Nuclear Officer) declined offers for a public meeting, but indicated a willingness to provide information if the Panel identified the need. The Panel did not identify a need for additional information from either Exelon or NEI to complete its review, which is summarized below and documented in the attached report.

At the request of the Panel, the Office of Nuclear Regulatory Research (RES) conducted risk analyses using the NRC's Standardized Plant Analysis Risk model for Byron Unit 1.[19] These analyses informed the Panel's response to the question from the EDO regarding the risk significance of the relevant accident sequences.

1.2 Proposed Compliance Backfit and Exelon Appeals

In the Backfit Letter, the NRC staff informed Exelon that it had determined that Byron and Braidwood are not in compliance with GDCs 15, 21, and 29; 10 CFR 50.34(b); and the plant-specific design bases that were expected to demonstrate there will be no progression of Category II events to Category III events. The NRC staff stated that based on its review of Byron and Braidwood UFSAR Sections 15.5.1, 15.5.2, and 15.6.1, the UFSAR predicts water discharge through a valve that is not "qualified" for water discharge. Therefore, the NRC staff concluded that the UFSAR does not contain analyses that demonstrate that the plants' structures, systems, and components (SSCs) will meet the design criteria for ANS Condition II faults as stated in Byron and Braidwood UFSAR Section 15.0.1.2. Based on the SE attached to its letter,[20] the NRC staff found that the licensee must take action to resolve the non-compliance.

The Backfit SE addressed three accident analyses in Chapter 15 of the Byron and Braidwood UFSAR: (1) IOECCS; (2) CVCS malfunction that increases reactor coolant inventory; and (3) inadvertent opening of a pressurizer safety or relief valve (IOPORV). The NRC staff noted that each ANS Condition II event must be shown to meet the following:

1. no fuel damage,
2. no overpressure of the reactor coolant system (RCS) or main steam system, and
3. no progression into an event of a more serious category without another independent fault.

Regarding an IOECCS, the NRC staff stated in Section 3.1.2.1 of the Backfit SE that use of the block valve to isolate a stuck-open PORV was unacceptable. The NRC staff stated that Westinghouse recommended this approach in 1993 and that the NRC staff rejected this approach in 2005 (RIS 2005-29[21]).

In Section 3.1.2.4 of the Backfit SE, the NRC staff stated that the Byron and Braidwood IOECCS analysis depends on water discharge through the PSVs. The NRC staff faulted the

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[19] NRC 2016f
[20] Referred to as the Backfit SE in the remainder of the report.
[21] NRC 2005b

licensee for "not appl[ying] the single-failure assumption" and stated that the following information was necessary to support water qualification of the PSVs:

1. In accordance with the American Society of Mechanical Engineers (ASME) *Boiler and Pressure Vessel Code* (BPV Code), Section III, provide the original Overpressure Protection Report defining operating conditions and required relief capacities, and manufacturer's certification and test results

2. In accordance with the ASME *Code for Operation and Maintenance of Nuclear Power Plants* (OM Code), provide inservice test history for PSVs, including water and steam tests, or provide correlation test for alternative test fluid.

Regarding a CVCS malfunction, the NRC staff stated in Section 3.2 of the Backfit SE that the licensee had not provided an analysis for the CVCS malfunction that increases reactor coolant inventory that demonstrates the plants' ability to meet the requirements of an ANS Condition II event.

Regarding an IOPORV, the NRC staff stated in Section 3.3 of the Backfit SE that the licensee had not provided an analysis for the IOPORV that extends long enough into the transient to demonstrate the event would not transition from an ANS Condition II event to an ANS Condition III event.

In the Backfit SE, the NRC staff referenced Millstone[22] and Callaway[23] license amendments as examples of licensees upgrading PORVs for water discharge; a Beaver Valley extended power uprate (EPU) license amendment[24] as an example of qualifying PORVs for water discharge; and Turkey Point[25] and St. Lucie Unit 2[26] EPU amendments as additional precedent in support of the backfit decision.

In the NRR Appeal, Exelon asserted that the NRC had not justified invoking the compliance exception to the backfit rule. Exelon stated that the NRC approved its IOECCS analysis in the Uprate SE and Setpoint SE.

In the NRR Appeal Decision, the NRC staff stated that the previous approvals were inconsistent with the Agency's general position on the known and established standard at issue, in this case the progression of ANS Condition II events to higher level events. The NRC staff stated that the fact that the NRC staff were aware of references to EPRI reports on the ability of these non-water qualified PSVs to reseat in certain circumstances is not sufficient to support the licensee's position.

In the EDO Appeal, Exelon stated that the NRC had misidentified the "known and established standard" at issue as the prohibition of ANS Condition II events progressing to ANS Condition III events. Exelon asserted that the standard in question concerns what is necessary to "qualify" valves for water discharge. Exelon contended that this standard is the EPRI testing and analysis, and that the NRC has agreed that Byron and Braidwood meet this standard. Exelon also contended that the change in NRC staff position on prior approvals is not a mistake of fact,

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[22] NRC 1998
[23] NRC 2000
[24] NRC 2006
[25] NRC 2012a
[26] NRC 2012b

but rather a new or modified interpretation of compliance with NRC requirements, for which use of the compliance exception provided for in the Backfit Rule is not appropriate.

1.3 Backfit Rule and the Compliance Exception

Backfitting is defined by 10 CFR 50.109(a) as:

> … the modification of or addition to systems, structures, components, or design of a facility; or the design approval or manufacturing license for a facility; or the procedures or organization required to design, construct or operate a facility; any of which may result from a new or amended provision in the Commission's regulations or the imposition of a regulatory staff position interpreting the Commission's regulations that is either new or different from a previously applicable staff position … .

Unless one of three specified exceptions apply, the NRC may impose a backfit only if it performs a backfit analysis in accordance with 10 CFR 50.109(a)(2) and determines in accordance with 10 CFR 50.109(a)(3) "that there is a substantial increase in the overall protection of the public health and safety or the common defense and security to be derived from the backfit and that the direct and indirect costs of implementation for that facility are justified in view of this increased protection."

Section 50.109(a)(4) sets forth the three exceptions to the requirements of 10 CFR 50.109(a)(2) and (a)(3). The first exception, the compliance exception, applies if the "modification is necessary to bring a facility into compliance with a license or the rules or orders of the Commission, or into conformance with written commitments by the licensee." The second and third exceptions relate to actions necessary to ensure adequate protection or to actions that involve defining or redefining adequate protection.

The Commission explained its intended application of the compliance exception in the Statements of Consideration (SOC) accompanying the 1985 final rule amending 10 CFR 50.109:[27]

> The compliance exception is intended to address situations in which the licensee has failed to meet known and established standards of the Commission because of omission or mistake of fact. It should be noted that new or modified interpretations of what constitutes compliance would not fall within the exception and would require a backfit analysis and application of the standard.

In the same SOC, the Commission acknowledged that staff interpretations of rules are not legally binding, but the Commission also stated that "staff interpretations of broadly stated rules are often necessary to give a rule effect and in some instances may be a causal factor in initiating a backfit."[28]

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[27] NRC 1985, at 38103

[28] NRC 1985, at 38102. The 1985 backfit rule was vacated by a Federal court on grounds unrelated to the compliance backfit exception. See *Union of Concerned Scientists v. U.S. Nuclear Regulatory Com'n*, 824 F.2d 108, 119-20 (1987). In 1988, the Commission amended the backfit rule (NRC 1988b) to address the court's concerns, but did not change the 1985 rule's compliance exception provision. Thus, the quoted statements from the 1985 rule are the applicable expression of Commission intent regarding compliance backfits.

By its terms, the compliance exception applies to actions necessary for compliance with rules, licenses, and orders, or for conformance with written commitments.[29] Also, the Commission explicitly acknowledged the importance of staff interpretations of rules in the regulatory process. Thus, the Panel understands the term "known and established standard" to include standards established in rules, licenses, orders, and written commitments, and NRC interpretations of rules. Some standards may be broad-based, while others may apply only to a limited number of plants. As stated in NUREG-1409, "[i]nformal or formal communications to one licensee are not official positions to all licensees. ... Orders, licenses, and written commitments are applicable only to a particular licensee."

The failure to meet a known and established standard is grounds for a compliance backfit if this failure is due to "omission or mistake of fact." Thus, if a licensee obtains NRC approval of an alternative to a specific standard set forth in guidance, that standard and guidance could not be used to support a compliance backfit unless the NRC's approval of the alternative was based on an omission or mistake of fact. "Known and established standards" are to be distinguished from "new or modified interpretations of what constitutes compliance," which do not fall within the compliance exception. The Panel understands the term "new or modified interpretations" to include situations where the NRC has, in effect, "changed its mind" on how to interpret the language of a requirement or on how much assurance is necessary to conclude that the requirement is met. Levels of assurance might be established in terms such as acceptable probabilities or consequences, conservative assumptions, or sufficient margin.

Additional background information on the Backfit Rule and the compliance exception is provided in Appendix A to this report.

1.4 A Brief History of Pressurizer Valve Issues

Appendix B to this report provides a summary of the NRC and industry's testing, evaluation, and other consideration of PORVs and PSVs since the TMI Unit 2 (TMI-2) accident in 1979. This historical review provides context for discussion of valve "qualification" in the Backfit SE. It also provides the basis for the Panel's conclusions regarding the "known and established standard" for "qualification" in the context of the TMI Action Plan item and subsequent activities, as well as how it should be interpreted in the Byron and Braidwood licensing basis.

In light of the NRC staff's assertion that the licensee had not applied the "single-failure assumption" as noted above, the Panel considered the applicability of the single failure criterion to PSVs. The Panel expended considerable effort in searching for an answer to what appears to be a simple question: "Are PSVs active components subject to the single failure criterion, or are they passive components exempt from it?" NRR staff have taken the position that PSVs have consistently been treated as active components.

In the Panel's evaluation of the treatment of PSV failure potential (Section 3 below), an historical perspective is provided. In general, the Panel found that the classification of a component as "active" or "passive" depends on its design, application, and function. For example, passive components almost always do not need external power; usually do not need an external actuator (e.g., signal)[30]; sometimes do not involve any mechanical motion (e.g., movement of a

[29] NUREG-1409 (NRC 1990c) defines written commitments broadly to include the "final safety analysis report, licensee event reports, and docketed correspondence, including responses to NRC bulletins, generic letters, inspection reports, or notices of violation and confirmatory action letters."
[30] For example, SECY-77-439 (NRC 1977) states: "Examples [of passive failures in fluid systems] include

valve disc)[31]; and sometimes do not involve any motion, either fluid or mechanical (e.g., piping). International Atomic Energy Agency (IAEA) TECDOC-1624 states that "[s]afety related terms such as passive and inherent safety have been widely used, particularly with respect to advanced nuclear plants, generally without definition and sometimes with definitions inconsistent with each other." This guidance further defines four level of "passivity" to "to help eliminate confusion and misuse of the terms by members of the nuclear community." In addition, SECY-05-0138[32] also acknowledges and discusses inconsistencies in the use and application of the term "passive."

The introduction to the GDCs and the related footnote define the applicability of the single failure criterion in terms of electrical versus fluid systems, and active versus passive components. Neither the GDCs nor NRC guidance define which characteristics of passive components are necessary to make a component exempt from the single failure criterion. Some examples are clear: pipes are passive components and pumps and motor-operated valves that operate to perform their safety functions are active components. As discussed in Section 3.6 below, check valves might be classified as active or passive components depending on specific considerations.

With respect to PSVs, the ASME BPV Code applicable to Byron and Braidwood includes requirements for overpressure protection that relate to the single failure criterion through several specific design and construction requirements. As a result, the PSVs are conservatively sized with sufficient margin to accommodate a single failure although the single failure criterion is almost never explicitly discussed or applied in accident analyses. The Byron and Braidwood UFSAR states that "adequate overpressurization protection is provided by the three installed safety valves." Neither the UFSAR system descriptions nor the safety analyses provide detailed discussions of potential PSV failures or their consequences. The principal discussion of potential PSV failures in the accident analyses occurs in the evaluation of an inadvertent opening of a PSV in UFSAR Section 15.6.1.

Most relevant for the current issue, the Byron and Braidwood UFSAR analyses of overpressure events (e.g., loss of load, loss of feedwater) do not apply the single failure criterion to cause a PSV to stick open (i.e., fail to reseat) when opening on steam flow. In addition, the UFSAR Feedwater System Pipe Break analysis (Chapter 15.2.8) does not apply the single failure criterion to cause a PSV to stick open either during steam discharge or during water discharge. A survey of other Westinghouse-designed plants showed that this treatment of PSV valve performance during anticipated operational occurrences (AOOs, similar to ANS Condition II events) and postulated accidents (similar to ANS Condition IV events) has been consistent and without any identified exceptions.[33]

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the failure of a simple check valve to move to its correct position when required, the leakage of fluid from failed components, such as pipes and valves—particularly through a failed seal at a valve or pump—or line blockage. Motor-operated valves which have the source of power locked out are allowed to be treated as passive components."

[31] For example, NUREG-1800 (NRC 2001c) states that "'[p]assive' structures and components, for the purpose of the license renewal rule, are those that perform an intended function … without moving parts or without a change in configuration or properties … 'passive' may also be interpreted to include structures and components that do not display 'a change of state.'"

[32] NRC 2005a

[33] Examples include Watts Bar (NRC 1982 and TVA 1983), North Anna (NRC 1976), and AP1000 (Westinghouse 2011).

1.5 History of Westinghouse NSAL and Related Activities

Appendix C to this report provides the Panel's review of the issues identified by Westinghouse in NSAL-93-13 and its Supplement 1, how various licensees responded to these issues, and how the NRC was involved in reviewing and approving these actions. This review provides the basis for the Panel's conclusions related to the approach taken by Byron and Braidwood to address these issues in their licensing basis, as well as on the "known and established standard" for event escalation from ANS Condition II to ANS Condition III, referred to hereafter as the "non-escalation position."

2 SUMMARY OF PANEL FINDINGS

For the reasons provided in Section 3, the Panel concludes that in 2001 and 2004 and at present, the known and established standard of the Commission is 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. The Panel also concludes that, in preparing the Uprate SE and the Setpoint SE, the NRC staff exercised reasonable and well-informed engineering judgment when the NRC staff concluded that the PSVs were unlikely to stick open. The non-escalation position does not establish specific standards for valve qualification, so the non-escalation position, standing alone, provides no basis for rejecting the licensee's reliance on EPRI valve testing. Moreover, the Panel finds that no mistake or error occurred in the licensee's or previous staff's reliance on the EPRI testing program that included an evaluation of water discharge through pressurizer valves.[34] Therefore, the Panel also concludes that the position on valve qualification in the Backfit SE is a new or modified interpretation of what constitutes compliance.

3 DISCUSSION

The compliance exception to the Backfit Rule is intended to address failures to meet known and established Commission standards because of omission or mistake of fact. New or modified interpretations of what constitutes compliance do not fall within the exception. The Panel reviewed and evaluated the information referenced in this report to determine if, in 2001 and 2004, there was a known and established standard of the Commission relating to the potential for PSVs to fail following water discharge during IOECCS events.

In addition, the Panel considered the issue of "known and established standards of the Commission" as it relates to "event escalation." In the NRR Appeal Decision, the NRC staff stated that the Backfit SE "showed that the approvals at issue for Braidwood and Byron were inconsistent with the Agency's general position on the known and established standard at issue, in this case the progression of [ANS] Condition II events." The Panel recognizes that the non-escalation position, although not included in NRC regulations, is widely referenced in reactor licensing bases as an approach for addressing AOOs and postulated accidents as articulated in the GDCs. The non-escalation position is incorporated in Section 15.0.1.2 of the Byron and Braidwood UFSAR as "By definition, these faults (or events) do not propagate to cause a more serious fault, i.e., [ANS] Condition III or IV events."

[34] "Pressurizer valves" is used in this report to refer to either PORVs or PSVs when discussing issues common to both types of valves.

Neither Exelon nor the Panel disputes that the non-escalation position is now, and was in 2001 and 2004, a part of the licensing basis of both Byron and Braidwood. The Panel supports the NRC staff's view that non-escalation (from ANS Condition II to ANS Condition III or IV) is a known and established standard applicable to Byron and Braidwood. However, the Panel agrees with Exelon that the fundamental issue is not the non-escalation position, but the appropriate standard for PSV water discharge. In the absence of a PSV failure to reseat, the concerns articulated in the backfit related to event classification, event escalation, and compliance with 10 CFR 50.34(b) and GDCs 15, 21, and 29 would no longer be at issue.

The Panel's evaluation of the treatment of PSV failure potential includes an assessment of multiple relevant references, which are discussed chronologically in the sections that follow.

3.1 General Design Criteria (1971)

In 1971, the Atomic Energy Commission published the GDCs, which had been under development since 1965.[35] The introduction to Appendix A addresses "Single Failure" in the section on Definitions and Explanations. The paragraph on single failures includes a footnote stating: "The conditions under which a single failure of a passive component in a fluid system should be considered in designing the system against a single failure are <u>under development</u>" (emphasis added).

3.2 Commission Paper on Single Failure (1977)

In response to several staff concerns and differing views on the subject of application of the single failure criterion, the Acting Director of NRR issued SECY-77-439 "[t]o inform the Commission of the present status and future use of the Single Failure Criterion as a tool in the reactor safety process."[36] In part, that paper addressed the application of the single failure criterion to passive components in fluid systems, stating that "[a]pplication of the [single failure] concept is complicated by the interrelationships between the various fluid and electrical systems and their supporting auxiliaries in a nuclear power plant. Furthermore, there is a need to stipulate the events and associated assumptions which must be considered during application of the Single Failure Criterion."

SECY-77-439 specifically spoke to how "additional passive failures"—that is, failures in addition to the initiating event—had been and should be addressed, stating (with emphases added):

> During subsequent years [since the single failure footnote quoted above was published] staff <u>assumptions regarding the nature of passive failures which should be considered have not been completely consistent</u> and there has been some disagreement. However, on the basis of the licensing review experience accumulated in the period since 1969, it has been judged <u>in most instances that the probability of most types of passive failures in fluid systems is sufficiently small that they need not be assumed</u> in addition to the initiating failure in application of the Single Failure Criterion to assure safety of a nuclear power plant.

SECY-77-439 provides definitions and examples for distinguishing between active and passive failures. Among these examples, SECY-77-439 cites "the failure of a simple check valve to

[35] AEC 1971
[36] NRC 1977

move to its correct position when required" as a passive failure. Of the examples cited in SECY-77-439, the check valve example is most similar from a mechanical perspective to the PSV failure addressed in the Backfit SE, as explained below in the discussion of SECY-94-084.

SECY-77-439 also stresses the use of engineering judgment relating to the probability of component failure and does not suggest that valve "certification" or "qualification" in accordance with ASME standards should be invoked as the basis for such decisions.

3.3 TMI Action Plan Item II.D.1 (1980)

As an element of the TMI Action Plan, the NRC staff required licensees to address the capability of relief and safety valves to perform their intended functions without failure. Specifically, Item II.D.1 states that "[p]ressurized-water reactor [PWR] and boiling-water reactor [BWR] licensees and applicants shall conduct testing to qualify the [RCS] relief and safety valves under expected operating conditions for design-basis transients and accidents." With reference to planned EPRI testing and other generic industry test programs, NUREG-0737 specified provisions for then-operating nuclear power plants and applicants for operating licenses and holders of construction permits to address the TMI Action Plan items, including Item II.D.1. NUREG-0737 stated, for the performance testing of relief and safety valves for Item II.D.1, that "[t]he testing should demonstrate that the valves will open and reclose under the expected flow conditions."

Although limited in scope, the EPRI test results did not identify any generic issues with PSVs or PORVs sticking open following water discharge. The NRC staff approvals summarized below show that the word "qualify" in this TMI Action Plan item was not intended to refer to ASME valve certification or qualification. Instead, "qualify" was used in a less formal sense to refer to a reasonable judgment that the valve would open to relieve pressure and then reliably reseat. As referenced in NUREG-0737, the EPRI test program was the widely used approach to address TMI Action Plan Item II.D.1 at PWR nuclear power plants. The Westinghouse Owners Group submitted WCAP-10105 to the NRC in 1982 to demonstrate the acceptability of the EPRI testing program for PSVs and PORVs in Westinghouse-designed PWRs.[37]

3.4 NRC Closure of TMI Action Plan Item II.D.1 for Byron and Braidwood (1988-1990)

A 1988 letter from the NRC staff to the licensee for Byron found the licensee's reliance on EPRI testing of PSVs to be acceptable.[38] The 1988 SE states that the test program was designed "[t]o reconfirm the integrity of the overpressure protection system and thereby assure that the [GDCs] are met." As discussed in Appendix B to this report, the 1988 SE describes the evaluation of the PSVs and PORVs for feedwater line break accidents that would include water discharge, and determined that the EPRI tests were applicable to the Byron and Braidwood PSVs and PORVs. Based on the NRC staff and contractor review, the 1988 SE found that the performance of the PSVs and PORVs was acceptable based on the EPRI tests.

For the specific extended high pressure injection event, the 1988 SE states that water discharge through the PSVs and PORVs could be disregarded because of the long time available for operator action. However, the SE addressed water discharge through the PSVs and PORVs as part of the feedwater line break evaluation.

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[37] WOG 1982
[38] NRC 1988c, referred to as the 1988 SE

In the cover letter for the 1988 SE, the NRC staff states that the licensee should develop and adopt plant procedures to inspect the pressurizer valves after each lift involving loop seal or water discharge. The 1988 SE contains no reference to or suggestion of a need for certification of these valves in accordance with the ASME BPV Code for water discharge capability. In 1990, the use of the EPRI test program was also found similarly acceptable for Braidwood.[39]

3.5 Westinghouse NSAL-93-013 and Supplement 1 (1993-1994)

In 1993, Westinghouse sent NSAL-93-013 to operating nuclear power plants in response to its discovery that potentially non-conservative assumptions had been used in the licensing analysis of the IOECCS event. Westinghouse recommended that licensees determine if their pressurizer safety relief valves (PSRVs)[40] "are capable of closing following discharge of subcooled water." Westinghouse noted that the PSRVs might have been designed or "qualified" to relieve subcooled water. Westinghouse also noted that "licensees may have qualified these valves in compliance to NUREG-0737, Item II.D.1." If the PSRVs were not designed or qualified for subcooled water discharge, Westinghouse recommended that licensees reevaluate the IOECCS event with three possible options of (1) reducing emergency core cooling system (ECCS) flow used in the safety analysis, (2) using a less restrictive operator response time, or (3) crediting the use of one or more PORVs to help mitigate the accident.

In Supplement 1 to NSAL-93-013, Westinghouse alerted licensees to potential reduced time for operator action if a positive displacement pump (a typical part of the CVCS) were in service, and to the need to qualify the PSRVs and the piping downstream of the PSRVs and PORVs if water discharge from the pressurizer is predicted.

Some licensees submitted license amendments that involved improvements to the PORVs and their circuitry to avoid water discharge through the PSVs (e.g., Salem[41], Millstone[42], Callaway[43], and Diablo Canyon[44]). The NRC staff review and approval of those proposed improvements relied on engineering judgment relative to the various test information and PORV circuitry upgrades described by individual licensees. The licensee for Byron and Braidwood submitted an LAR for similar PORV improvements,[45] but that request was later withdrawn.[46]

As indicated below, the Panel's sampling review found two plants, in addition to Byron and Braidwood, that chose to address this issue by crediting the capability of PSVs to relieve water, based on the EPRI testing performed in response to TMI Action Plan Item II.D.1.

3.6 Commission Paper on Passive Plant Designs (1994)

In 1994, in preparation for the design certification reviews of passive reactor designs (e.g., the Westinghouse Advanced Passive 1000 (AP1000) and the General Electric Economic Simplified

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[39] NRC 1990a
[40] Westinghouse used the term PSRVs. The specific valves for Byron and Braidwood should be designated as "safety valves" or "pressurizer safety valves" as they are by the manufacturer, in the ASME BPV Code, and by the licensee. This difference in terminology is not significant to any of the findings or conclusions in this report.
[41] NRC 1997
[42] NRC 1998
[43] NRC 2000
[44] NRC 2004a
[45] ComEd 1998
[46] ComEd 1999

Boiling-Water Reactor (ESBWR)), the NRC staff presented nine issues to the Commission for policy decisions.[47] Although PSV categorization and performance requirements were not explicitly addressed, the paper does include an issue on "Definition of Passive Failure" and an extensive discussion on whether check valves are passive or active components and how they should be addressed in current plants and future passive designs.

SECY-94-084 recognizes the GDCs and SECY-77-439 as establishing long-standing requirements and guidance in this area. The paper acknowledges that the industry (including EPRI documents and ANSI/ANS 58.9[48]) have been inconsistent with respect to check valve failures, sometimes considering them as "active failures" and sometimes as "passive failures." In SECY-77-439, however, the NRC staff stated that the failure of a simple check valve to move to its correct position when required was a "passive failure." In addition, SECY-94-084 states that "[i]n licensing reviews, however, only on a long-term basis [e.g., long-term recirculation cooling following a loss of coolant accident (LOCA)] does the NRC staff consider passive failures in fluid systems as potential accident initiators in addition to initiating events." The paper also states that "[f]or current plants, the NRC staff normally treats check valves, except for those in containment isolation systems, as passive devices during transients or design-basis accidents."

Furthermore, SECY-94-084 states that "[r]edefining check valves as active components, subject to consideration for single active failures would cause these valves to be evaluated in a more stringent manner than that used in previous licensing reviews" (emphasis added). The NRC staff then recommended (and the Commission agreed[49]) that the NRC staff should "maintain the current licensing practice for passive component failures on the passive [advanced light water reactor] ALWR designs, and to redefine check valves, except for those whose proper function can be demonstrated and documented, in the passive safety systems as active components subject to single failure consideration." Therefore, the NRC's position on check valves was changed only for passive ALWR designs going forward.

The Panel considers the opening function of check valves and PSVs to be similar in that they both open through the motion of the valve disk under differential pressure with no external signal or motive power. The Panel also recognizes that the ambiguity with respect to "passive" versus "active" component definitions and nomenclature exists for safety valves. In addition, the passive or active classification of check valves or safety valves may differ based on design considerations, inservice testing, or accident analyses. For example, the PSVs and PORVs as well as numerous check valves are classified as active components in the Byron and Braidwood inservice testing programs. However, for purposes of applying the single failure criterion in the GDC context, the Panel concludes that it is appropriate to consider the potential failure of a PSV following water discharge as a passive failure, consistent with the treatment of check valve failures for the operating fleet.

3.7 Draft Standard Review Plan Revision (1996)

The 1996 draft revision to SRP Section 15.5.1 – 15.5.2 on IOECCS and CVCS malfunctions includes extensive updates to the 1981 revision, but neither version includes any discussion, criteria, or guidance on applying ASME Code requirements to PSVs or on applying the single failure criterion or any other failure assumption to PSVs.[50]

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[47] NRC 1994a
[48] ANS 1981
[49] NRC 1994b
[50] NRC 1996

3.8 Power Uprate Reviews and License Amendments (2001-2006)

As part of the 2001 power uprate review for Byron and Braidwood, the NRC staff approved the analysis of an IOECCS (UFSAR Section 15.5.1) that included pressurizer filling, PSV water discharge, ECCS termination, and PSV closure. In the Backfit SE, the NRC staff indicates that the 2001 license amendment was predicated on the NRC's mistaken (unsubstantiated) belief that the valves were ASME-qualified (certified). However, a review of the SE and associated RAIs shows that, in 2001, the NRC staff was well aware of the nature of the EPRI testing that the licensee relied on. The Panel did not find any evidence that the licensee claimed or the NRC staff believed that the valves were "qualified" in an ASME certification sense; rather, the record shows thorough consideration of the testing conducted on valves of the type installed at the plant and a technical judgment that this testing provided appropriate qualification.

The Panel's conclusion was confirmed via discussions with the individual who was the responsible Section Chief in the Reactor Systems Branch at the time. He informed the Panel that the 2001 license amendment was based on the exercise of staff engineering judgment and there was no discussion of ASME certification or qualification of valves. In addition, the Panel found that the NRC approved power uprates for other nuclear power plants that included staff evaluation of water discharge through PORVs or PSVs based on test information provided by individual licensees. For example, in 2001, the NRC granted a power uprate for Shearon Harris that included the operability of PORVs and PSVs during the discharge of subcooled water, referencing TMI Action Plan Item II.D.1.[51] As noted above, in 2006, the NRC also granted a power uprate for Beaver Valley. The SE for this Beaver Valley amendment referred to RIS 2005-29 and found reasonable assurance that the PSVs would adequately discharge water and reseat following a spurious safety injection actuation, based on the EPRI test data from 1981 and an evaluation of the temperature of the liquid being discharged.

During the NRC evaluations of license amendments since the TMI-2 accident, the NRC staff has specified in some SEs that a PORV or PSV would be assumed to stick open if it was not qualified for liquid service. To address this concern, the NRC staff reviewed and accepted a variety of test information (including EPRI, Wyle, and vendor testing) submitted by individual licensees to demonstrate the capability of PORVs or PSVs to reseat following water discharge. In the sample of SEs it reviewed, the Panel did not find a specific requirement for the PORVs or PSVs to be certified under the ASME BPV Code as capable of passing water and reclosing.

In 2004, the NRC issued license amendments for Byron and Braidwood granting an adjustment to the PSV setpoints. In an RAI, the NRC staff requested that the licensee perform a quantitative analysis regarding the number of opening cycles during which the PSV would be expected to pass water and the temperature of the water being discharged. In the Setpoint SE, the NRC staff concluded that the analysis was acceptable for assuring that the PSVs would remain operable following a spurious safety injection event.

[51] NRC 2001d

3.9 RIS 2005-29 (2005)

In 2005, the NRC staff issued RIS 2005-29 "to notify licensees of a concern identified during recent reviews of power uprate [LARs]." The RIS addressed the manner in which some licensees acted in response to NSAL-93-013. The RIS was issued at the division level in NRR and does not include a record of office-level concurrence. The RIS was not reviewed by CRGR. Although no documentation was readily available regarding the CRGR's decision not to review, it appears that the lack of a CRGR review stemmed from the assertions in the RIS such as these:

- "This RIS requires no action or written response and, therefore, is not a backfit under 10 CFR 50.109. Consequently, the NRC staff did not perform a backfit analysis."
- "This RIS is informational and pertains to a NRC staff position that does not depart from current regulatory requirements and practice."

A key statement in RIS 2005-29 is the following (with emphasis added):

> The NRC staff's position is noted in the power uprate review standard, as follows: "For the [IOECCS] and [CVCS] malfunctions that increase reactor coolant inventory events: (a) non-safety-grade pressure-operated relief valves should not be credited for event mitigation and (b) pressurizer level should not be allowed to reach a pressurizer water-solid condition.".

However, the cited review standard (RS-001), which is explicitly limited to EPUs, states that "[t]he staff does not intend to impose the criteria and/or guidance in this review standard on plants whose design bases do not include these criteria and/or guidance. No backfitting is intended or approved in connection with the issuance of this review standard."[52]

This intent of RS-001 to define and clarify the scope of EPU reviews, but not impose new requirements or new interpretations of requirements, was confirmed by the Panel in discussions with the manager responsible for developing and issuing RS-001. Therefore, contrary to the RIS statement, neither RS-001 nor RIS 2005-29 documented "known and established standards of the Commission" applicable to Byron and Braidwood.

The Panel also notes that neither RIS 2005-29 nor its draft Revision 1,[53] which is currently under development, discuss water discharge certification requirements in accordance with the ASME BPV Code. In fact, as stated above, the NRC issued a 2006 power uprate amendment for Beaver Valley in which the SE cited RIS 2005-29 and yet relied on the EPRI testing data to address the concern.

3.10 SECY-05-0138 (2005)

SECY-05-0138 presents a comprehensive history of the application of the single failure criterion, including extensive discussion of the treatment of passive components in fluid systems.[54] The paper enclosed a July 2005 draft of a technical report on the single failure criterion. Section 4.2.2 of this report acknowledges that "[o]ne particular issue identified in this

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[52] NRC 2003
[53] NRC 2015a
[54] NRC 2005a

project is the continued existence of the footnote to the definition of single failure in 10 CFR [Part] 50 Appendix A stating that the regulatory position on considering passive failures in fluid systems is under development." In Section 2.5.3, the draft report quotes from SECY-77-439 (discussed above) and recognizes that in current practice, as in 1977, "[p]assive failures in fluid systems are generally excluded from single-failure assessments."

SECY-05-0138 and the accompanying draft report present three alternatives for using a risk-informed and performance-based approach to address the single failure issue. The draft report clarifies that all of the alternatives "could include developing a position on single passive failures in fluid systems to replace the footnote now in 10 CFR Part 50 Appendix A definitions."

These documents make it clear that, with few exceptions, neither the NRC staff nor the Commission has established specific requirements relating to the treatment of passive component failures in fluid systems. The Panel believes the existence of this Commission paper, contemporaneous with discussions on potential PSV failures (e.g., RIS 2005-29), makes it clear that no specific "known and established standards" on PSV failures had been developed between 1977 and the time of the Byron and Braidwood license amendments in 2001 and 2004.

3.11 Standard Review Plan Revision (2007)

Revision 2 to SRP Section 15.5.1 – 15.5.2 states:

> If the plant is equipped with PORVs that are (1) safety-related equipment and (2) qualified for water relief, then they may be assumed to reseat properly after having relieved water. The ~~that "[t]he pressurizer safety valves~~[PSVs], too, may be assumed to reseat properly after having relieved water; but only if such valves have been qualified for water relief.~~"~~

However, this section does not reference ASME BPV Code requirements for safety valve certification.

3.12 Backfit Letter and Subsequent Appeals (2015-2016)

The Backfit SE is predicated on the following positions:

- "water relief through a valve that is not qualified for water relief will cause that valve to stick in its fully open position" (emphasis added)
- "the licensee ... has not applied the single-failure assumption" (emphasis added)
- "nor have they provided ASME water qualification documentation for the PSVs ... the ASME ... original Overpressure Protection Report ... inservice test history ... including both water and steam tests" (emphasis added)

The Backfit SE argues that an IOECCS would escalate to a more severe event. Such an escalation would be contrary to the Byron and Braidwood licensing basis (i.e., contrary to the ANS non-escalation position) and could be in non-compliance with the GDCs (as included in the Byron and Braidwood licensing basis) since an IOECCS with a stuck-open valve had not been analyzed and shown to meet the appropriate criteria for an AOO.

Based on its review of all the relevant documents and discussions with the individuals (staff and managers) involved in the original review and the backfit, the Panel has developed an understanding of the regulatory requirements and practices, the potential safety issues, and backfit rule obligations. The Panel has determined that the numerous, complex, and detailed regulatory and technical issues all depend on the answers to two critical questions on valve performance:

- Must the PSVs in question be assumed to fail given liquid water discharge because of the lack of ASME certification for water discharge?
- Must the PSVs be assumed to fail in accordance with the GDC "single failure" requirements?

In the Backfit SE, the NRC staff indicates that "[o]ne assumption that is particularly important to the non-escalation criteria is that water relief through a valve that is not qualified for water relief <u>will cause</u> that valve to stick in its fully open position" (emphasis added). The Panel concludes that this issue—the treatment of potential valve failure—is not only "particularly important," it is the critical issue upon which the compliance backfit hinges.

Based on the historical evidence, the Panel concludes that there is not now, nor has there been, a known and established Commission standard (1) that PSVs must be assumed to fail following water discharge in the absence of ASME certification for water discharge, or (2) that PSVs must be assumed to fail as part of single failure criterion analysis. The NRC staff's determination that ASME certification is necessary first appears in the Backfit SE. The determination that application of the single failure criterion is necessary first appears in the draft Revision 1 to RIS 2005-29. The Panel has not identified these positions being stated in any final NRC guidance document.

The Panel also concludes that in 2001 and 2004 and at present, the known and established standard of the Commission is 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. In preparing the Uprate SE and the Setpoint SE, the NRC staff exercised reasonable and well-informed engineering judgment when the NRC staff concluded that the PSVs were unlikely to stick open. The NRC staff reviewers involved in the 2001 power uprate review were among the most experienced and senior reviewers in their areas of expertise. The NRC staff valve expert involved in the review was the agency's most knowledgeable individual on PSVs and the relevant ASME Code requirements, and was a nationally recognized expert. The Panel did not find any evidence that the NRC staff's issuance of the 2001 or 2004 license amendments was based on an omission or mistake of fact. Rather, the current NRC staff positions on valve qualification in the Backfit SE are new or modified interpretations of compliance.

In interactions with the Panel, NRR staff emphasized several issues raised in the Backfit Letter. The Panel summarizes its consideration of those issues in the following subsections.

3.12.1 Non-Escalation Position and Valve Failure

In the Backfit SE, the NRC staff discussed the definition of event conditions in ANS-51.1/N18.2-1973 and the provision in this standard that events of one condition do not propagate to cause a more serious fault (non-escalation position). In interactions with the Panel, NRR staff provided several clarifications on this topic, summarized by the Panel as follows:

- ANS-51.1/N18.2-1973 defines the categories of design basis transients and accidents based on an anticipated frequency of occurrence (annually for ANS Condition II events).
- It is a long-standing NRC position that escalation from one condition to another is not acceptable.
- ANS-51.1/N18.2-1973 constitutes a known and established standard that has been reflected in NRC guidance documents and in the licensing basis of each U.S. nuclear power plant.

The Panel confirmed that this ANS standard is referenced in several places in Chapter 15 of the Byron and Braidwood UFSAR. The Panel agrees that the non-escalation position is an established standard applicable to Byron and Braidwood, but did not identify historical evidence that implementation of this standard requires Exelon to assume that its pressurizer valves will fail open under water discharge conditions, to apply the single failure criterion to PSV failure in these circumstances, or to impose ASME Code requirements for certification, qualification, or testing of PSVs for water discharge.

3.12.2 Non-Escalation Position and Return to Service

In the Backfit SE, the NRC staff makes reference to the time it would take to clean up a contaminated containment following a stuck-open pressurizer valve. In interactions with the Panel, NRR staff re-emphasized concerns that extended steam and water discharge through the pressurizer valves would result in the failure of the pressurizer relief tank rupture disk, would require repair of the damaged PSVs, and might cause an extended time period for the return to service of the nuclear power plant.

The Panel does not consider the time period necessary for the licensee to perform radioactive clean-up activities in the containment building, to inspect and conduct any necessary repairs to the PSVs, or to prepare for plant startup, to constitute issues that support a compliance backfit imposed by the NRC. The NRC staff and inspectors would verify that these activities are conducted appropriately to protect the public health and safety prior to plant restart. The Backfit SE states that UFSAR Section 15.5.1.3 "implie[s]" that the plant will return to operation in a "short period," but the Panel sees no support for a timing requirement in UFSAR Section 15.5.1.3. Also, the Panel has not identified a regulatory interest in limiting the time needed for the plant to return to operation.

3.12.3 TMI Action Plan Item II.D.1 and EPRI Testing

Although the Backfit Letter and NRR Appeal Decision do not speak explicitly to TMI Action Plan Item II.D.I, in interactions with the Panel, NRR staff stated that the known and established standard in question is the TMI Action Plan Item II.D.I standard for licensees and applicants to conduct testing to qualify the RCS relief and safety valves under expected operating conditions for design-basis transients and accidents. As discussed above and in Appendix B to this report, the NRC accepted the EPRI testing to satisfy TMI Action Plan Item II.D.1 for Byron and Braidwood in SEs forwarded by letters in 1988 and 1990. Therefore, the Panel considers this known and established standard referenced by the NRC staff to have been met for Byron and Braidwood.

In interactions with the Panel, the NRR staff further stated that an omission or mistake of fact occurred when the licensee failed to acknowledge that the EPRI testing program did not evaluate water discharge from the pressurizer valves during extended high pressure safety injection for Byron and Braidwood. As discussed in Appendix B to this report, in the 1988 and 1990 SEs on the Byron and Braidwood response to TMI Action Plan Item II.D.1, the NRC staff evaluated the capability of the PSVs and PORVs during feedwater line break accidents, including water discharge. In these SEs, the NRC staff found that the performance of the PSVs and PORVs with water discharge was acceptable based on the EPRI tests. Therefore, the Panel does not agree that the licensee's reference to the EPRI testing program was an omission or mistake of fact.

3.12.4 ASME Code Certification

In the Backfit SE, the NRC staff stated that certain ASME Code information would be necessary to support water qualification of the PSVs. In interactions with the Panel, NRR staff stated that, to satisfy the standard for water discharge capability of pressurizer valves, it would be necessary to conduct flow capacity certification in accordance with the ASME BPV Code and inservice testing throughout the service life in accordance with the ASME OM Code. The NRR staff referenced certain licensing actions in which water discharge was not considered acceptable, or different actions were required.[55]

As discussed in Appendix C to this report, the NRC staff required additional actions for some licensees to support reliance on the PORVs for water discharge and to avoid water discharge through the PSVs. The Panel found, however, that the NRC staff also allowed some licensees to rely only on EPRI testing without significant additional activities. The Panel did not identify instances where the NRC staff imposed certification by the ASME BPV Code and testing in accordance with the OM Code, or required alternatives to the ASME BPV or OM Codes, in the examples of NRC staff review of water discharge capability for pressurizer valves.

In interactions with the Panel, the NRR staff also identified specific ASME Code provisions that it viewed as supporting the position that ASME Code requirements apply to qualification of pressurizer valves for water discharge. The NRR staff, however, did not provide evidence that these provisions have consistently been interpreted as the NRC staff is now interpreting them. Given the NRC's treatment of TMI Action Plan Item II.D.1 and the NRC staff's historical licensing practice, the Panel concludes that the NRR staff's current application of the ASME Code is not supported by the historical record.

3.12.5 Conduct of 2001 and 2004 Reviews

In light of the wide range of NRC staff positions during the review of pressurizer valve capability since the TMI-2 accident, the Panel agrees that, in the course of preparing the 2001 Uprate SE or Setpoint SE, the NRC staff could have considered the need for the licensee for Byron and Braidwood to improve the reliability of the PSVs or PORVs for water discharge or to avoid water discharge through the PSVs by PORV improvements. The NRC staff may have been able to justify additional actions, but they determined that it was not necessary. Instead, the NRC staff reviewers in 2001 used their expert engineering judgement to determine that it was not necessary to assume that the PSVs or PORVs would stick open with water discharge, based on EPRI test information, licensee supplemental information, and their own technical experience.

In discussions with the Panel, NRR staff raised a concern that the Setpoint SE does not document a re-review of the qualification of the PSVs and noted that if the Uprate SE had not found water discharge through the PSVs to be acceptable, it is unlikely that the NRC staff would

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[55] Salem (NRC 1997), Millstone (NRC 1998), and Callaway (NRC 2000)

have approved this 2004 amendment. In Appendix C to this report, the Panel summarizes the discussion in the Setpoint SE of the PSV water discharge capability. The Panel recognizes that a staff review may rely on a previous more extensive review to determine the acceptability of a similar request. The Panel does not consider the review approach used in 2004 to challenge the adequacy of the 2001 review.

4 RESPONSE TO THE EDO QUESTIONS

In establishing the Panel, the EDO asked the Panel to answer five specific questions, as well as evaluating the overall appropriateness of the backfit. The answers to these questions are provided below.

4.1 Were the approvals based on a mistake? If so, what was the mistake and what are the implications for ~~Byron and~~ Braidwood and Byron?

In responding the question, the Panel has considered the differing views of the NRR staff and the licensee on this issue. Those positions are summarized below:

- In the NRR Appeal Decision, the NRC staff claims that "[t]he NRC erred in approving a sequence of events that allowed the [IOECCS], [CVCS] malfunction, and inadvertent opening of a pressurizer safety or relief valve analyses in the 2001 and 2004 [SEs]" and "the NRC staff understood the PSVs to be qualified for water relief when, in fact, they were not."

- Exelon claims in the NRR Backfit Appeal that "the compliance exception requires more than simply asserting that the prior staff approvals were wrong—the NRC must demonstrate that the prior approvals were erroneous because of an omission or mistake of fact at the time of the approval. The NRC has not made that case here."

The Panel concludes that, in 2001 and 2004, the NRC staff did not misunderstand the qualification status of the PSVs and that it was not a mistake to undertake a review of or make a technically based safety finding on the likely successful performance of the valves. In the Panel's opinion, the actions of the Reactor Systems Branch in 2001 to reach out to the Division of Engineering's Mechanical Engineering Branch for expert technical review assistance was both appropriate and commendable. The NRC staff reviewers involved in the 2001 power uprate review were among the most experienced and senior reviewers in their areas of expertise. The valve expert involved in the review was the NRC's most knowledgeable individual on PSVs and the relevant ASME Code requirements, and was a nationally recognized expert. The Panel cannot agree that the NRC staff was misinformed, ill-informed, or in error, or that it made incorrect or inappropriate decisions.

4.2 What is the known and established standard for water qualification of PSVs?

The Panel concludes 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. No more detailed or prescriptive standard has been promulgated by the Commission.

4.3 What is the known and established standard for progression of postulated events between categories of severity?

For Byron and Braidwood, the standard for progression of postulated events between categories of severity is set forth in the UFSAR as described above. The Panel supports the NRC staff's view that non-escalation (from ANS Condition II to ANS Condition III or IV) is a known and established standard applicable to Byron and Braidwood. This issue of event escalation is also a focus of RIS 2005-29 and the draft Revision 1 to RIS 2005-29 that was issued for public comment in 2015. The Panel concludes that the IOECCS (an AOO per the GDC definition and an ANS Condition II event) would escalate to a more severe event if a PSV were to stick open, or if both a PORV stuck open and its block valve failed to close. Such an escalation would be contrary to the Byron and Braidwood licensing basis (i.e., contrary to the ANS non-escalation position) and could be in non-compliance with the GDC (as included in the Byron and Braidwood licensing basis), since an IOECCS with a stuck-open valve had not been analyzed and shown to meet ~~to~~ the appropriate criteria for an AOO. However, this event progression standard does not establish specific standards for valve qualification to determine whether a valve would stick open and cause this escalation. Therefore, it is not the basis for a compliance backfit given the current set of facts.

4.4 Does the current licensing basis for Braidwood and Byron ~~and Braidwood~~ comply with the applicable regulations? Is it adequate to provide protection to public health and safety?

The Panel concludes that Byron and Braidwood do comply with the applicable regulations based on the UFSAR analyses, which the NRC staff found acceptable through a reasonable and technically sound evaluation using appropriate Commission safety standards.

4.5 Given that Exelon suggests that the NRC pursue a cost-justified substantial safety enhancement backfit, what is the contribution to overall plant risk of the current configuration at ~~Byron and~~ Braidwood and Byron?

The Panel requested RES to provide information and insights on the risk significance of the sequence at issue, to assure that the Panel's judgments were being made with a full understanding of their significance, and to assist in responding to the EDO question.

The RES study suggests that the most significant IOECCS sequence, assuming that all pressurizer overfill events lead to a small LOCA, contributes approximately 1 percent of the total internal event core damage frequency (CDF). In its report, RES estimated a maximum benefit (CDF reduction) from a "perfect backfit" (i.e., always preventing pressurizer overfill) of 1.5E-07 per year. If the PSVs are not assumed to always fail following water discharge (consistent with the NRC staff expert judgment in 2001) or a smaller improvement than a "perfect backfit" were considered, the risk-reduction benefit of implementing the backfit would be even smaller.

The Panel is aware of and sensitive to two important issues related to this question. First, NRR, not the ~~appeal~~ Panel, is responsible for any decisions on alternative application of the backfit rule to this issue (through the other categories of adequate protection or cost-justified substantial safety enhancement). Second, the Panel does not wish to imply that "the contribution to plant risk" should be seen as the only measure of enhanced safety. The issues of event classification and the non-escalation of events are essentially defense-in-depth concepts. Defense in depth has a recognized role and value in the regulatory process. The Panel is also aware that not every defense-in-depth feature has the same safety significance, and that the estimated risk significance (measured in core damage frequency) is very relevant.

Within the context described above, the Panel concludes that the contribution to overall plant risk is very small.

5 SUMMARY AND CONCLUSIONS

The compliance exception to the Backfit Rule is intended to address failures to meet known and established Commission standards because of omission or mistake of fact. New or modified interpretations of what constitutes compliance do not fall within the exception. Therefore, to address the appeal of the proposed compliance backfit, the Panel focused on determining if this case is most appropriately characterized as one in which the licensee "failed to meet known and established standards of the Commission because of omission or mistake of fact," or rather as a case of a "new or modified interpretations of what constitutes compliance."

The NRC staff's compliance backfit argument depends on two separate determinations:

1. the assumed failure of PSVs to reclose after passing water, and

2. the necessity of preventing "event escalation" (i.e., the position that "an incident of moderate frequency should not generate a more serious plant condition without other faults occurring independently").

For the NRC staff's compliance backfit conclusion to be valid, both of these determinations must meet the above compliance backfit standard by involving failure to meet known and established standards of the Commission.

In the first of these determinations, the NRC staff's compliance backfit is based on the assumption in the Backfit SE that the PSV fails to reclose given the absence of "ASME water qualification documentation." As indicated in the Backfit SE, the Uprate SE involved a technical evaluation of safety valve capability and likely performance under water-discharge conditions rather than a simple assumption of a failure. The NRR Appeal Decision indicates that "the 2001 and 2004 approvals occurred because the NRC staff understood the PSVs to be qualified for water relief when, in fact, they were not."

The Panel carefully considered these views and has reviewed the relevant documents including the licensee's responses to the NRC staff's RAIs,[56] the technical branch's SE input,[57] and the Uprate SE. 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 certification sense; rather, the record shows thorough consideration of the testing conducted on valves of the type installed at the plant and a technical judgment that this testing provided appropriate qualification.

On the basis of its review, the Panel concluded that the NRC staff who prepared the Uprate SE did <u>not</u> misunderstand the qualification status of the PSVs and that it was <u>not</u> a mistake to undertake a review of or make a technically based safety finding on the likely successful performance of the valves. In the Panel's opinion, the actions of the Reactor Systems Branch in 2001 to reach out to the Division of Engineering's Mechanical Engineering Branch for expert

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[56] ComEd 2000b, Exelon 2001
[57] NRC 2001a

technical review assistance was both appropriate and commendable. The NRC staff reviewers involved in the 2001 power uprate review were among the most experienced and senior reviewers in their areas of expertise. The NRC staff valve expert involved in the review was the agency's most knowledgeable individual on PSVs and the relevant ASME Code requirements, and was a nationally recognized expert. The Panel disagreed that the NRC staff was misinformed, ill-informed, or in error, or that it made incorrect or inappropriate decisions.

The Panel concluded that three related technical and regulatory positions related to the PSVs (separate from the issue of the non-escalation position) underpin the backfit:

1. ASME water qualification (certification) documentation is required if a valve is to be assumed to reclose after passing water.

2. Water discharge through a steam-qualified valve will cause that valve to stick in its fully open position.

3. PSVs are subject to a single-failure assumption.

None of these positions were "known and established standards of the Commission" in 2001 or 2004 for determining when it was appropriate to assume a failure of PSVs to reseat. In fact, they were not "known and established standards of the Commission" in 2005 (when RIS 2005-29 was issued) or 2006 (when the Beaver Valley EPU was approved) or 2007 (when Revision 2 to SRP Section 15.~~1~~5.1 – 15.~~1~~5.2 was issued).

Moreover, these positions do not appear to be "established standards of the Commission" at present. The 2007 version of SRP Section 15.~~1~~5.1 – 15.~~1~~5.2 allows credit for PORVs and PSVs if they have been "qualified for water relief." The NRC staff's determination that ASME certification is necessary first appears in the Backfit SE and is not addressed in any final NRC guidance document. The determination that application of the single failure criterion is necessary first appears in the draft Revision 1 to RIS 2005-29 and is not included in any final NRC guidance document.

The Panel 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. In earlier documents addressing this topic, beginning with NUREG-0737, the use of the word "qualified" or "qualification" implies a general demonstration of capability, such as in the EPRI testing done in response to TMI Action Plan Item II.D.1. In light of this standard, the Panel concluded that, when preparing the Uprate SE and the Setpoint SE, the NRC staff exercised reasonable and well-informed engineering judgment to conclude that the PSVs were unlikely to stick open.

Overall, the Panel concluded that the NRC staff's position on valve qualification in the Backfit SE is a new or modified interpretation of what constitutes compliance in addressing potential PSV failures following water discharge. Although this new staff position represents a well-intentioned and conservative approach that could provide additional safety margin, it does not provide a basis for a compliance backfit.

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.

The Panel's findings, therefore, support the Exelon backfit appeal.

6 ADDITIONAL PANEL THOUGHTS

In addition to the specific finding relating to the backfit appeal, the Panel believes it is important to acknowledge, and for the NRC staff and licensees to appreciate, that water discharge through a PSV not specifically designed for such service is undesirable and should be minimized or avoided as a matter of conservative engineering and prudent operations. This is reinforced by the information provided in NSAL-93-013 and its Supplement 1, and the actions by various licensees in response to these documents, as well as the limited scope of the EPRI testing conducted over 30 years ago.

Operator training, control room procedures to terminate the event before pressurizer filling, and use of PORVs rather than reliance on PSVs, are clearly preferred and prudent measures, whether they form the facilities' UFSAR licensing basis and are assumed in the accident analyses or not.

The PSVs in question were designed for steam service. Steam relief is their normal service condition and applies to their ASME BPV Code certification. The Panel supports the previous NRC staff determinations for Byron and Braidwood and certain other plants that PSVs experiencing water discharge during an abnormal or accident condition need not be assumed to fail since there was a reasonable and technically well-informed engineering judgement to the contrary. However, the Panel also considers the actions by various licensees to improve the reliability and performance of the PORVs to avoid water discharge through the PSVs to be prudent in light of the design specifications of the PSVs.

The Panel considered but could not determine the extent to which the licensee for Byron and Braidwood addressed crediting water discharge through the PSVs, PORVs, or PORV block valves in the Byron and Braidwood inservice testing programs. The Panel recognizes that the difference between the intended use of these valves for overpressure protection and their infrequent use in response to certain plant events might be considered in implementing appropriate inservice testing activities.

The Panel notes that water discharge through various pressurizer valves is not a new issue because water discharge has always been credited (by the licensee for Byron and Braidwood and other licensees) for the feedwater line break analysis in UFSAR Section 15.2.8.

APPENDIX A: HISTORY OF THE BACKFIT RULE AND THE COMPLIANCE EXCEPTION

The Backfit Rule

Title 10 of the *Code of Federal Regulations* (10 CFR), Section 50.109, "Backfitting," was originally promulgated in 1970.[58] Because of perceived deficiencies in the rule, the U.S. Nuclear Regulatory Commission (NRC) substantially revised it in 1985.[59] The 1985 rule was challenged in court, and the U.S. Circuit Court for the District of Columbia (D.C. Circuit) vacated this rule in its entirety. The D.C. Circuit took this action because it concluded that the revised rule could be interpreted to allow the NRC to consider costs in defining or redefining what is required for adequate protection of the public health and safety.[60] In response, the NRC revised the Backfit Rule in 1988 to remove any implication that costs could be considered in defining or redefining adequate protection.[61] The 1988 revisions only differed from the 1985 rule to the extent necessary to address the court's concerns. The 1988 rule was also challenged in court, but this time the D.C. Circuit upheld the rule.[62]

In its current form, 10 CFR 50.109(a)(1) defines backfitting as

> … the modification of or addition to systems, structures, components, or design of a facility; or the design approval or manufacturing license for a facility; or the procedures or organization required to design, construct or operate a facility; any of which may result from a new or amended provision in the Commission's regulations or the imposition of a regulatory staff position interpreting the Commission's regulations that is either new or different from a previously applicable staff position … .

Unless one of three specified exceptions apply, the NRC may impose a backfit only if it performs a backfit analysis in accordance with 10 CFR 50.109(a)(2) and determines in accordance with 10 CFR 50.109(a)(3) "that there is a substantial increase in the overall protection of the public health and safety or the common defense and security to be derived from the backfit and that the direct and indirect costs of implementation for that facility are justified in view of this increased protection."

Section 50.109(a)(4) sets forth the three exceptions to the requirements of 10 CFR 50.109(a)(2) and (a)(3). The first exception, the compliance exception, applies if the "modification is necessary to bring a facility into compliance with a license or the rules or orders of the Commission, or into conformance with written commitments by the licensee." 10 CFR 50.109(a)(4)(i). The second and third exceptions relate to actions ensuring adequate protection or to actions that involve defining or redefining adequate protection. 10 CFR 50.109(a)(4)(ii)-(iii).

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[58] AEC 1970 (Author and year citations in footnotes refer to the designation of references in Appendix D to this report.)
[59] NRC 1985
[60] *Union of Concerned Scientists v. U.S. Nuclear Regulatory Com'n*, 824 F.2d 108, 119-20 (1987).
[61] NRC 1988b
[62] *Union of Concerned Scientists v. U.S. Nuclear Regulatory Com'n*, 880 F.2d 552 (1989).

Commission Policy

The Commission addressed its intended application of the compliance exception in the 1985 rulemaking:[63]

> The compliance exception is intended to address situations in which the licensee has failed to meet known and established standards of the Commission because of omission or mistake of fact. It should be noted that new or modified interpretations of what constitutes compliance would not fall within the exception and would require a backfit analysis and application of the standard.

In the 1985 rule, the Commission acknowledged that staff interpretations of regulations are not legally binding, but the Commission also stated that "staff interpretations of broadly stated rules are often necessary to give a rule effect and in some instances may be a causal factor in initiating a backfit."[64] The Commission also stated, "Many of the most important changes in plant design, construction, operation, organization, and training have been put in place at a level of detail that is expressed in staff guidance documents which interpret the intent of broad, generally worked [sic] regulations."[65]

Backfitting Guidance

Extensive information regarding the appropriate implementation of backfitting is provided in ~~the~~ NUREG-1409.[66] Relevant excerpts from this guidance are provided below.

Applicable Regulatory Staff Positions

According to NUREG-1409, to be a backfit, "a new or revised staff position or requirement must be involved, that is, there must be a change in content or applicability of the previously applicable regulatory staff position (in the direction of increased safety requirements) … ." An applicable regulatory staff position is a requirement or position already specifically imposed on or committed to by a licensee. Examples of applicable regulatory staff positions include:

- legal requirements, as in explicit regulations, orders, and plant licenses and in amendments, conditions, and technical specifications
- written licensee commitments such as those contained in the final safety analysis report, licensee event reports, and docketed correspondence, including responses to NRC bulletins, generic letters, inspection reports, or notices of violation and confirmatory action letters
- NRC staff positions that are documented explicit interpretations of more general regulations and are contained in documents such as the Standard Review Plan, branch technical positions, regulatory guides, generic letters, and bulletins

A similar list of examples is provided in Manual Chapter 0514,[67], which is also included as Appendix D to NUREG-1409. Manual Chapter 0514 was referenced in the 1988 rulemaking,

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[63] NRC 1985, at 38103
[64] *Id.* at 38102
[65] *Id.* at 38103. The 1988 rulemaking neither revised the compliance exception as stated in the 1985 rule nor provided additional guidance on its interpretation.
[66] NRC 1990c

and a working draft was provided to the Commission for information in SECY-88-102.[68] Manual Chapter 0514 provides a definition of "applicable regulatory staff positions" that is slightly more detailed than the definition in NUREG-1409. This definition from Manual Chapter 0514 is quoted below, with additional detail beyond NUREG-1409 emphasized in underlined text.

> Applicable regulatory staff positions are those already specifically imposed upon or committed to by a licensee at the time of the identification of a plant-specific backfit, and are of several different types and sources:
>
> a. Legal requirements such as in explicit regulations, orders, plant licenses (amendments, conditions, technical specifications). Note that some regulations have update features built in, as for example, 10 CFR 50.55a, Codes and Standards. Such update requirements are applicable as described in the regulation.
>
> b. Written commitments such as contained in the [Final Safety Analysis Report], [Licensee Event Reports], and docketed correspondence, including responses to Bulletins, responses to Generic Letters, Confirmatory Action Letters, responses to Inspection Reports, or responses to Notices of Violation.
>
> c. NRC staff positions[69] that are documented, approved, explicit interpretations of the more general regulations, and are contained in documents such as the [Standard Review Plan], Branch Technical Positions, Regulatory Guides, Generic Letters, and Bulletins; and to which a licensee or an applicant has previously committed to or relied upon. Positions contained in these documents are not considered applicable staff positions to the extent that staff has, in a previous licensing or inspection action, tacitly or explicitly excepted the licensee from part or all of the position.[70]

How Regulatory Positions are Established

NUREG-1409 provides responses to a number of questions regarding backfitting. The following response was given to questions asking, "Is it appropriate for the NRC staff to rely on informal or formal communications to other licensees as official NRC positions? What about NRC tacit approval of documents?"

> Informal or formal communications to one licensee are not official positions to all licensees. Section 053 of Manual Chapter 0514 identifies what can be applied as official staff positions in a plant-specific context. They are legal requirements such as contained in explicit regulations, orders, and plant licenses; written commitments such as contained in final safety analysis reports, licenses event reports, and docketed correspondence; and documented, approved explicit interpretations such as contained in the [Standard Review Plan], branch technical positions, regulatory guides, generic letters, and bulletins. Orders, licenses, and written commitments are applicable only to a particular licensee.

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[67] NRC 1988c

[68] NRC 1988a

[69] Requirements may be imposed by rule or order. Staff interpretations such as examples of acceptable ways to meet requirements are not requirements in and of themselves.

[70] Imposition of a staff position from which a licensee has previously been excepted is a backfit.

If the NRC staff previously exempted a licensee from a legal requirement or approved position, it is not applicable to that licensee for the purpose of backfit consideration. Explicit exemption would be done formally in writing. The Appendix to NRC Manual Chapter 0514 discusses tacit approval under reanalysis of issues. Two situations are covered. In the first case, staff review of a previously accepted licensee action or program may result in a requested change. This would be classified as a backfit because it represents a change in a previous staff position and would require a backfit analysis (or a documented evaluation if it meets one of the exceptions listed in the backfit rule). In the second case, a licensee submittal committing to a specific course of action that has not received timely NRC staff review is implemented by the licensee. In this case, it is considered that the NRC staff tacitly accepted the licensee's action since timely notice to the contrary was not given. If the NRC staff subsequently adopts a different position and requests a change in the licensee action, this change may be classified as a backfit and thus require a backfit analysis (or a documented evaluation if it meets one of the exceptions listed in the backfit rule).

NUREG-1409 also addresses a question regarding tacit approvals by an inspector: "If an inspector has previously accepted (i.e., provided tacit approval of) a licensee's method, does a specific request for change constitute a backfit and if so, is a backfit analysis required?" The response is:

> Cases where an inspector provides tacit approval are relatively rare. Simply not challenging a licensee's practice normally would not be considered tacit approval. The only example provided in Manual Chapter 0514 is a case where the NRC has indicated tacit approval by not acting in a reasonable time on a licensee submittal and the licensee has moved ahead to implement the proposal described in the submittal. For the purpose of this question, it would most likely arise in connection with review of a licensee response to an inspection report.
>
> Explicit approval could be provided in an inspection report that states that a particular approach is acceptable. However, conclusions of that nature are usually made in [safety evaluations] rather than inspection reports.

Compliance Backfit Guidance

NUREG-1409 gives the following response to the question, "[h]ow does the backfit rule apply to new staff positions that reflect an evolving understanding of technical issues?"

> An evolving understanding of issues does not, by itself, define which category fits a particular backfit. Judgment must be applied to the facts of each particular case to determine whether the backfit is for compliance, to provide adequate protection, to redefine adequate protection, or to achieve a cost-justified substantial safety enhancement. For example, with regard to compliance, the 1985 statement of considerations for 10 CFR 50.109 indicates that "the compliance exception is intended to address situations where the licensee has failed to meet known and established standards of the Commission because of omission or mistake of fact....new or modified interpretations of what constitutes compliance would not fall within the exception...."

NUREG-1409 also provides an example where an evolving understanding of technical issues resulted in a compliance backfit that was apparently justified for at least some licensees. In

response to industry claims that Bulletin 88-11[71] lacked any backfitting justification, the NRC staff responded:

> Although the justification was not printed in the bulletin, NRC Bulletin 88-11, "Pressurizer Surge Line Thermal Stratification," was justified as a backfit. It is an example of a backfit that was determined by the responsible NRC official to be required as a matter of compliance with existing requirements and commitments. The CRGR reviewed the bulletin and concurred. The regulations currently require licensees to meet the applicable codes of the American Society of Mechanical Engineers (ASME), *Boiler and Pressure Vessel Code.* Because of the NRC staff's concern with the integrity of the surge line, licensees were requested to perform their fatigue analysis in accordance with the latest ASME Section III requirements that incorporate high cycle fatigue analysis. The justification provided by the NRC staff was that previously unconsidered thermal stratification phenomenon may invalidate the existing analysis performed to confirm the integrity of the surge line.
>
> Subsequently, it was understood that some licensees believed that the NRC staff's rationale was in error because they were not committed to the latest ASME Section III requirements by virtue of their license commitment. However, the issue became moot because these licensees undertook the analysis voluntarily in view of the safety importance of the issue and the fact that previous versions of the ASME Code did not completely address the concern.

[71] NRC 1988e

APPENDIX B: QUALIFICATION OF PRESSURE RELIEF VALVES IN NUCLEAR POWER PLANTS IN RESPONSE TO TMI-2 ACCIDENT

Byron and Braidwood Design and Code Requirements

Nuclear power plants in the United States use various types of pressure relief valves to protect personnel and equipment from overpressure events within reactor fluid systems. Pressure relief valves include safety valves, safety relief valves, and relief valves, with different designs, operating conditions, and requirements. The American Society of Mechanical Engineers (ASME) *Boiler and Pressure Vessel Code* (BPV Code), Section III, Division 1, specifies requirements for the design, operation, installation, and testing of pressure relief valves used for various functions in nuclear power plants.[72] For example, the ASME BPV Code (2007 Edition) in Article NB-7000, Overpressure Protection, specifies requirements several service conditions:

- steam and air or gas service for safety valves;
- steam, air or gas, and liquid service for safety relief valves;
- liquid service for relief valves; and
- steam, air or gas, and liquid service for pilot operated or power actuated pressure relief valves.

The ASME *Code for Operation and Maintenance of Nuclear Power Plants* (OM Code) provides requirements for the preservice and inservice testing (IST) programs for pressure relief valves in nuclear power plants.

Braidwood, Units 1 and 2 (Braidwood) and Byron, Units 1 and 2 (Byron) are Westinghouse-designed pressurized-water reactors (PWRs) that received their construction permits under Title 10 of the *Code of Federal Regulations* (10 CFR), Part 50, in December 1975. The pressurizer for each unit is equipped with three pressurizer safety valves (PSVs) and two power-operated relief valves (PORVs). The three PSVs are Crosby Model HP-BP-86, size 6M6 (6-inch), spring-~~-~~loaded pop type, opened by direct fluid pressure. The PORVs are Copes-Vulcan Model D-100-160 3-inch pneumatic-actuated globe valves that respond to a signal from the pressure sensing system or to manual control. Each PORV can be isolated by a motor-operated block valve.

The ASME BPV Code of record for the design of the PSVs at Byron and Braidwood ~~was~~ is the 1971 Edition through the Winter 1972 addenda of the ASME BPV Code, Section III. The ASME BPV Code applicable to Byron and Braidwood includes requirements for overpressure protection, including the following:~~. For example,~~

Section NB-7300, "Overpressure Protection Report," in NB-7320(f) requires that the report include the redundancy and independence of the pressure-relief devices and their associated pressure-sensing and controls systems employed to preclude a loss of overpressure protection in the event of a failure of any pressure-relief device, or its sensing element, or its associated control, or an external power source.

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[72] References to individual ASME Code publications are not provided in Appendix D, but they are publicly available from ASME for a fee.

Paragraph NB-7411, "Relieving Capacity of Pressure-Relief Devices," specifies that the total rated relieving capacity shall be sufficient to prevent a rise in pressure of more than 10 percent~~%~~ above system design pressure (at design temperature) within the pressure-retaining boundary of the system, under any pressure transient anticipated to arise as summarized in the Overpressure Protection Report.

Paragraph NB-7421, "Required Number and Capacity of Pressure-Relief Devices for Nuclear Systems," states that the required relieving capacity intended for overpressure protection of a nuclear power system or portions of the system shall be secured by the use of at least two pressure-relief devices.

At the time of the Byron and Braidwood operating license review, Revision 1 of ~~NRC~~ Standard Review Plan (SRP)~~, Revision 1 (July 1981), Chapter~~ Section 15.5.1-15.5.2~~, "Inadvertent Operation of ECCS and Chemical and Volume Control System Malfunction that Increases Reactor Coolant Inventory," and Chapter~~ and Section 15.6.1~~, "Inadvertent Opening of a PWR Pressurizer Pressure Relief Valve or a BWR [boiling-water reactor] Pressure Relief Valve,"~~ provided general staff guidance for these plant transients.[73] In March 2007, the NRC staff issued Revision 2 to these SRP sections with significantly more detail, including a statement that PSVs and PORVs are assumed to fail open if they relieve water without being qualified.[74]

Actions Following Three Mile Island, Unit 2 Accident

The accident at Three Mile Island, Unit 2 (TMI-2) on March 28, 1979, included failure of a PORV on the pressurizer to reclose properly during the event. Based on lessons learned from the TMI-2 accident, the NRC issued recommendations regarding performance testing of safety and relief valves used in nuclear power plants in NUREG-0578~~ (July 1979), "TMI-2 Lessons Learned Task Force Status Report and Short-Term Recommendations."~~.[75] In particular, the NRC staff recommended in Section 2.1.2, "Performance Testing for BWR [boiling-water reactor] and PWR Relief and Safety Valves," of NUREG-0578 that nuclear power plant licensees commit to provide performance verification by full-scale prototypical testing for all relief and safety valves.

~~On October 31,~~In October 1980, the NRC issued a letter to all then-operating nuclear power plants and applicants for operating licenses and holders of construction permits forwarding NUREG-0737.[76] ~~, "Clarification of TMI Action Plan Requirements." Requirement~~TMI Action Plan Item II.D.1~~, "Performance Testing of Boiling-Water Reactor and Pressurized-Water Reactor Relief and Safety Valves (NUREG-0578, Section 2.1.2),"~~ in NUREG-0737 specified the NRC position that PWR and BWR licensees and applicants shall conduct testing to "qualify" the reactor coolant system (RCS) relief and safety valves under expected operating conditions for design-basis transients and accidents. The detailed clarification in NUREG-0737 of this NRC position specified the following:

> Licensees and applicants shall determine the expected valve operating conditions through the use of analyses of accidents and anticipated operational occurrences referenced in Regulatory Guide 1.70, Revision 2. The single failures

[73] NRC 1981b and NRC 1981c
[74] NRC 2007b and NRC 2007c
[75] NRC 1979a
[76] NRC 1980b and NRC 1980c

applied to these analyses shall be chosen so that the dynamic forces on the safety and relief valves are maximized. Test-pressures shall be the highest predicted by conventional safety analysis procedures. [RCS] relief and safety valve qualification shall include qualification of associated control circuitry, piping, and supports, as well as the valves themselves.

A. Performance Testing of Relief and Safety Valves--The following information must be provided in report form by October 1, 1981:

(1) Evidence supported by test of safety and relief valve functionability for expected operating and accident (non-~~ATWS~~[anticipated transient without scram]) conditions must be provided to NRC. The testing should demonstrate that the valves will open and reclose under the expected flow conditions.

(2) Since it is not planned to test all valves on all plants, each licensee must submit to NRC a correlation or other evidence to substantiate that the valves tested in the EPRI (Electric Power Research Institute) or other generic test program demonstrate the functionability of as-installed primary relief and safety valves. This correlation must show that the test conditions used are equivalent to expected operating and accident conditions as prescribed in the final safety analysis report (FSAR). The effect of as-built relief and safety valve discharge piping on valve operability must also be accounted for, if it is different from the generic test loop piping.

(3) Test data including criteria for success and failure of valves tested must be provided for NRC staff review and evaluation. These test data should include data that would permit plant-specific evaluation of discharge piping and supports that are not directly tested.

In describing the type of review to be conducted for this regulatory position, the NRC staff stated the following:

Pre-implementation review will be performed for EPRI and BWR test programs with respect to qualification of relief and safety valves. Also, the applicants' proposal for functional testing or qualification of PWR valves will be reviewed. Post-implementation review will also be performed of the test data and test results as applied to plant-specific situations.

In specifying the documentation required to satisfy this regulatory position, the NRC staff stated the following:

Pre-implementation review will be based on EPRI, BWR, and applicant submittals with regard to the various test programs. These submittals should be made on a timely basis as noted below, to allow for adequate review and to ensure that the following valve qualification dates can be met:

Final PWR (EPRI) Test Program--July 1, 1980

Final BWR Test Program--October 1, 1980

Block Valve Qualification Program--January 1, 1981

Post-implementation review will be based on the applicants' plant-specific submittals for qualification of safety relief valves and block valves. To properly evaluate these plant-specific applications, the test data and results of the various programs will also be required by the following dates:

PWR (EPRI)/BWR Generic Test Program Results--July 1, 1981

Plant-specific submittals confirming adequacy of safety and relief valves based on licensee/applicant preliminary review of generic test program results--July 1, 1981

Plant-specific reports for safety and relief valve qualification--October 1, 1981

Plant-specific submittals for piping and support evaluations--January 1, 1982

Plant-specific submittals for block valve qualification--July 1, 1982~~.~~

~~In a letter dated July 27, 1982, to the NRC staff, the Westinghouse Owners Group (WOG) submitted WCAP-10105 (June 1982), "Review of Pressurizer Safety Valve Performance as Observed in the EPRI Safety and Relief Valve Test Program." In WCAP-10105, the WOG indicated that the design specification for PSVs in Westinghouse-designed nuclear power plants is for steam service only. Based on a review of the EPRI test data, the WOG concluded that the valves performed with chatter, but did not identify any valve damage. (ADAMS LL Accession No. 8208190310, Microfiche 14387:191-301)~~

EPRI Testing

In October 1982, EPRI issued NP-2670-LD to address testing of PORVs.[77] This report has been referenced by certain licensees (e.g., Section 15.2.14 of the North Anna, Units 1 and 2 Updated Final Safety Analysis Report (UFSAR)[78]).

In December 1982, EPRI issued NP-2628-SR~~, "EPRI PWR Safety and Relief Valve Test Program – Safety and Relief Valve Test Report," that~~, which described safety and relief valve tests for types of valves in service at nuclear power plants.[79] In particular, Section 3.5 ~~in EPRI NP-2628-NP discusses~~documented the testing of Crosby safety valves similar to the PSVs at Byron and Braidwood, including two water tests. The report indicated chattering of the safety valves with subsequent inspection finding galled surfaces and damage to internal parts. Section 4.6 ~~in EPRI NP-2628 discussed~~documented testing of Copes-Vulcan relief valves similar to the pressurizer PORVs at Byron and Braidwood, although the extent of water testing ~~is~~ was not fully described. The report indicated no damage found during the inspection of the Copes-Vulcan relief valves. The report did not indicate any failures of the Crosby or Copes-Vulcan valves to reseat after discharging water during the testing. ~~(ADAMS LL Accession No. 8407130197, Microfiche 25588:082-262)~~

~~In January 1983,~~ EPRI also ~~issued~~published NP-2770-LD in the early 1980s to describe~~, "EPRI/C-E PWR Safety Valve Test Report," that described~~ the testing of PWR primary system

[77] EPRI 1982a
[78] VEPCO 2015
[79] EPRI 1982b

safety valves. Volume 1, issued in December 1982, provides a summary of the test program and its results.[80] Section 4.5 of Volume 1 indicates that the following tests were performed on the Crosby 6M6 PSV: 11 steam tests with filled loop seals, 3 steam-to-water transition tests, and 2 water tests. The report states that the valve experienced chatter during the tests, and one water test had to be terminated. The individual volumes of EPRI NP-2770-LD discuss the test results for each specific PSV type. Volume 6, issued in March 1983, provides the test details for the Crosby 6M6 PSV.

Westinghouse Evaluation of EPRI Testing

In July 1982, the Westinghouse Owners Group (WOG) submitted WCAP-10105.[81] In WCAP-10105, the WOG indicated that the design specification for PSVs in Westinghouse-designed nuclear power plants is for steam service only. Based on a review of the EPRI test data, the WOG concluded that the valves performed with chatter, but did not identify any valve damage.

In January 1988, Westinghouse issued WCAP-11677, which compared the EPRI test data with feedwater line break safety analyses.[82] Westinghouse determined that all nuclear power plants addressed in the EPRI testing had PSVs that would operate reliably during water discharge. Westinghouse evaluated the performance of the Crosby 6M6 PSVs during the EPRI tests, and considered that the performance involved less significant flutter (half lift motion) than the chatter (full lift motion) determined in the EPRI report. Westinghouse concluded that the Crosby 6M6 PSV can pass slightly subcooled water at a minimum up to three times without damage.

Byron and Braidwood Licensing and Response to TMI Requirements ~~(EPRI NP-2770-LD, Volume 1, was obtained as a public document from the EPRI website. EPRI NP-2770-LD, Volume 6, could not be located within ADAMS or the NRC Record Retention Files, but is available for a fee from EPRI.)~~

~~In October 1982, EPRI issued NP-2670-LD, "EPRI/Wyle Power-Operated Relief Valve Phase III Test Report," to address testing of PORVs. This document could not be located in ADAMS despite its reference by nuclear power plant licensees. See, for example, North Anna Units 1 and 2 UFSAR (Revision 51, dated September 30, 2015), Section 15.2.14, "Spurious Operation of the Safety Injection System at Power."~~

The NRC ~~review~~ safety evaluation reports (SERs) associated with the issuance of the operating licenses ~~applications~~ for Byron and Braidwood included evaluation of the TMI Action Plan items.[83] ~~as discussed in the NRC Safety Evaluation Report (SER) for Braidwood Units 1 and 2, NUREG-1002, Section 1.1, "Introduction."~~ In ~~this SER section~~the introduction to the Braidwood SER, the NRC staff stated that the review and evaluation of compliance by the applicant with the licensing requirements established in NUREG-0660[84]~~, "NRC Action Plan Developed as a Result of the TMI-2 Accident,"~~ and TMI Action Plan Item II.D.1 were incorporated into the reviews summarized throughout the SER. ~~The NRC SER for Byron Units 1 and 2, NUREG-0876, also includes discussions of the NRC staff review of the TMI Action Plan items.~~

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[80] EPRI 1982c

[81] WOG 1982

[82] Westinghouse 1988

[83] NRC 1983 and NRC 1986b (Braidwood), NRC 1984 and NRC 1987a (Byron)

[84] NRC 1980a

Appendix E, "Requirements Resulting from TMI-2 Accident," to the Byron and Braidwood UFSAR in Section E.23, "Relief and Safety Valve Test Requirements (II.D.1)," references the 1982 transmittal from Consumers Power of a test report for the EPRI safety and relief valve test program~~indicated that a letter dated April 1, 1982, from D. Hoffman (Consumers Power) transmitted the Safety and Relief Valve Test Report for the EPRI PWR Safety and Relief Valve Test Program~~.[85] The UFSAR state~~s~~d that the final evaluation of the data indicated that the relief and safety valves will perform their intended functions for all expected fluid inlet conditions. The UFSAR also ~~indicated that the plant-specific final evaluation confirming the~~references the October 1982 licensee evaluation of the adequacy of the relief and safety valves ~~had been submitted by a letter from T. Tramm, dated October 26, 1982~~that had been submitted to the NRC.[86]

In Supplement 1 to the Braidwood SER ~~(NUREG-1002, Supplement 1, September 1986)~~,[87] in Section 3.9.3.3, "Design and Installation of Pressure Relief Devices," the NRC staff stated that EPRI had completed a full-scale valve testing program and referenced the July 1982 submittal of WCAP-10105~~, and that the WOG submitted the test results in WCAP-10105 in a letter dated July 27, 1982, from O. Kinglsey to S. Chilk. (ADAMS LL Accession No. 8208190307, Microfiche 14387:189-301)~~. The NRC staff stated that the applicant responded to a requirement to demonstrate operability of these valves through submittals dated July 1, 1982, ~~and~~ October 26, 1982, and December 30, 1983. On the basis of a preliminary review, the NRC staff concluded that the applicant's general approach to responding to this item was acceptable, and provided adequate assurance that the RCS overpressure protection systems at Braidwood ~~can~~ could adequately perform their intended functions. The NRC staff stated that if the detailed review revealed that modifications or adjustments to safety valves, PORVs, PORV block valves, or associated piping, ~~were~~ would be needed to ensure that all intended design margins were present, the NRC staff would require that the applicant make appropriate modifications. The NRC staff categorized this issue as a Confirmatory Item. ~~In Supplement 5 to the Byron SER (NUREG-0876, Supplement 5, October 1984) in Section 3.9.3.3, the NRC staff provided a similar discussion of the status of the NRC review of the capability of the Byron pressurizer valves. In Supplement 8 to the Byron SER (March 1987), the NRC staff stated TMI Action Plan Item II.D.I (3.9.3.3) had been closed in Supplement 5 to the Byron SER.~~ The NRC issued operating licenses for all four Byron and Braidwood Units between February 1985 and May 1988~~Byron Unit 1 in February 1985 and Unit 2 in January 1987, and Braidwood Unit 1 in July 1987 and Unit 2 in May 1988~~.

Closure of TMI Action Plan Item II.D.1 for Byron and Braidwood

Following the issuance of the ~~Byron and Braidwood~~ operating licenses, the NRC staff documented its review of the response to TMI Action Plan Item II.D.1 for Byron and Braidwood via two letters that transmitted similar ~~provided a letter dated August 18, 1988, from L. Olshan to H. Bliss, indicating that~~Technical Evaluation Reports (TERs) developed by Idaho National Engineering Laboratory (INEL).[88] In its letters, t~~Technical Evaluation Report (TER) EGG-NTA-8028 (January 1988) provided the review of the Byron response to TMI Action Plan Item II.D.1. (ADAMS LL Accession No. 8808260355, Microfiche 46653:240-269) T~~he NRC staff indicated that the licensee should develop and adopt plant procedures to inspect the pressurizer valves

[85] Consumers 1982

[86] ComEd 1982

[87] NRC 1986b. Similar discussion appears in NRC 1984 for Byron, and NRC 1987a (also for Byron) states that TMI Action Plan Item II.D.1 had been closed in NRC 1984.

[88] NRC 1988c (Byron) and NRC 1990a (Braidwood)

after each lift involving loop seal or water discharge. The TERs described the INEL review of the EPRI testing of ~~a~~ PSVs and PORVs similar to the Byron and Braidwood pressurizer valves. The TERs concluded that Byron and Braidwood had provided an acceptable response to TMI Action Plan Item II.D.1.

Section 4.2.3, "Extended High Pressure Injection [HPI] Event," of the TERs stated that the potential for water discharge in extended HPI events can be disregarded for an extended high pressure injection event because at least 20 minutes would be available for operator action.

Water discharge was evaluated, however, in ~~However,~~ Section 4.2.2, "FSAR Liquid Transients," of the TERs ~~of the TER~~This section discussed the evaluation of the PSVs and PORVs for feedwater line break accidents that would include water discharge, and determined that the EPRI tests were applicable to the Byron and ~~/~~Braidwood PSVs and PORVs.

In addition, Section 4.3.1, "Safety Valves," and Section 4.3.2, "Power Operated Relief Valves," of the TERs determined that the performance of the PSVs and PORVs was acceptable based on the EPRI tests, including water discharge tests. The TERs indicated that the PSV had two applicable tests: a loop seal /steam-water transition test where the valve opened, chattered and stabilized to close; and a saturated water test where the valve opened with water, chattered, and stabilized. The TERs indicated that the PORV opened and closed on demand in the loop seal /steam-water transition test, with a bending moment that was evaluated by analysis.

~~The TER concluded that Byron provided an acceptable response to TMI Action Plan Item II.D.1. On May 21, 1990, the NRC staff provided a letter from S. Sands to T. Kovach with the Braidwood TER that included similar findings. (ADAMS LL Accession No. 9005290209, Microfiche 53927:301-330)~~

~~In January 1988, WCAP-11677, "Pressurizer Safety Relief Valve for Water Discharge During a Feedwater Line Break," provided a description of the WOG comparison of the EPRI test data with feedwater line break safety analyses. This report was submitted as an attachment to a response to a request for additional information (RAI) dated May 8, 1989, from the licensee of the Seabrook nuclear power plant. (ADAMS Microfiche 49775:336 – 49756:017) As discussed in the report, the WOG determined that all nuclear power plants addressed in the EPRI testing have PSVs that will operate reliably during water discharge. The WOG evaluated the performance of the Crosby 6M6 PSVs during the EPRI tests, and considered that the performance involved less significant flutter (half lift motion) than the chatter (full lift motion) determined in the EPRI report. The WOG concluded that the Crosby 6M6 PSV can pass slightly subcooled water at a minimum up to three times without damage.~~

APPENDIX C: CONCERNS REGARDING PERFORMANCE OF PRESSURIZER VALVES UNDER WATER FLOW CONDITIONS

Westinghouse Nuclear Safety Advisory Letter

In 1993 and 1994, Westinghouse issued Nuclear Safety Advisory Letter (NSAL) 93-013 ~~(June 30, 1993) and NSAL-93-013,~~its Supplement 1 ~~(October 28, 1994)~~ to operating nuclear power plants (including Byron and Braidwood).[89] These advisories resulted from Westinghouse's discovery that potentially nonconservative assumptions were used in the licensing analysis of the Inadvertent Operation of the Emergency Core Cooling System at Power (IOECCS) event.

In NSAL-93-013, Westinghouse recommended that licensees determine ~~if~~ whether their pressurizer safety relief valves (PSRVs) are capable of closing following discharge of subcooled water. Westinghouse noted that the PSRVs might have been designed or "qualified" to relieve subcooled water. Westinghouse indicated that water discharge through the power-operated relief valves (PORVs) is not a concern, because the PORV block valves can be used to isolate the PORVs if they fail to close. If the PSRVs are not designed or qualified for subcooled water discharge, Westinghouse recommended that licensees re-evaluate the IOECCS event with three possible options of (1) reducing emergency core cooling system (ECCS) flow used in the safety analysis, (2) using a less restrictive operator response time, or (3) crediting the use of one or more PORVs to help mitigate the event.

In Supplement 1 to NSAL-93-013, Westinghouse informed licensees of a potential reduced time for operator action if a positive displacement pump is in service. Westinghouse also advised licensees~~, and to the need~~ to qualify the PSRVs and the piping downstream of the PSRVs and PORVs if water discharge from the pressurizer ~~is~~ were predicted.

Some licensees of operating nuclear power plants informed the NRC of their actions to address the potential concerns regarding ~~liquid service~~water discharge from ~~for~~ pressurizer safety valves (PSVs) and PORVs. A sample of actions by nuclear power plant licensees is summarized below in the "Plant-Specific Actions" section.

Additional NRC Generic Communications and Guidance

In ~~December~~ 2003, the NRC staff issued a review standard for extended power uprate (EPU) reviews~~NRR Review Standard for Extended Power Uprates (RS-001, Rev. 0)~~.[90] Item 8 on page 7 of the review standard states that pressurizer level should not be allowed to reach a pressurizer water-solid condition.

~~On December 14, 2005~~In 2005, the NRC issued Regulatory Issue Summary (RIS) 2005-29~~, "Anticipated Transients that could Develop into More Serious Events,"~~ to notify nuclear power plant licensees of a concern identified during ~~recent~~ reviews of power uprate ~~LARs~~requests.[91] In RIS 2005-29, the NRC staff stated that typically ~~ANS~~ Condition II ~~event~~ scenarios[92] involve discharging water through relief or safety valves that are not qualified for water discharge. The NRC staff stated that these valves are then assumed to fail in the open position and create a

[89] Westinghouse 1993 and Westinghouse 1994
[90] NRC 2003
[91] NRC 2005b
[92] As defined in American Nuclear Society (ANS) Standard 51.1/N18.2-1973 (ANS 1973).

small-break ~~LOCA~~loss-of-coolant accident (LOCA). The NRC staff stated that it was concerned that some licensees may be crediting PORVs without qualification for water discharge and without establishing additional restrictions to ensure the availability of PORVs and block valves. The NRC staff stated that the advice in Westinghouse NSAL-93-013 ~~allowing~~ to use the PORV block valves to isolate the PORVs is inconsistent with non-escalation position.

In ~~proposed~~ draft Revision 1 to RIS 2005-29, the NRC staff addresses the specific ANS Condition II scenarios of chemical volume and control system (CVCS) malfunction, inadvertent opening of a PORV or PSV (IOPSRV), and the IOECCS ~~event~~ event~~, and inadvertent opening of a PORV or PSV~~.[93] Regarding the CVCS malfunction, the NRC staff states that performing only ~~the~~ a reactivity anomaly analysis or assuming that this malfunction is not as severe as the IOECCS event is not acceptable. Regarding the IOPSRV event, the NRC staff stated that inadvertent opening of PSV or PORV could continue as an ANS Condition III small break LOCA and fails to meet the non-escalation position. Regarding the IOECCS event, the NRC staff states that five of the alternative approaches in NSAL-93-013 fail to meet the non-escalation position. The NRC staff indicated that these unacceptable alternative approaches are: ~~(1)~~

1. closing the block valve

2. ~~, (2)~~ assuming that the PORV is not operable

3. ~~, (3)~~ addressing a stuck-open PORV or PSV as a separate ANS Condition II event

4. ~~, (4)~~ determining that a stuck-open PORV or PSV is not as severe as a small break LOCA

5. ~~, and (5)~~ determining that RCS loss through PORV is made up by ECCS flow~~. Regarding inadvertent opening of PORV or PSV, the NRC staff states that inadvertent opening of PSV or PORV could continue as an ANS Condition III small break LOCA and fails to meet the non-escalation position.~~

Additional General PSV/PORV Information

In ~~August~~ 2004, EPRI issued Technical Report 1011047~~, "Probability of Safety Valve Failure-to-Reseat Following Steam and Liquid Relief - Quantitative Expert Elicitation,"~~, which evaluated the potential increase in failure rates following steam and liquid relief through safety valves based on expert judgement.[94] 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.

In ~~March~~ 2011, the NRC ~~published~~ summarized relief valve performance data in NUREG/CR-7037, ~~"Industry Performance of Relief Valves at U.S. Commercial Nuclear Power Plants through 2007,"~~ based on a study by the Idaho National Laboratory.[95] With respect to pressurizer PORVs, the report found four separate ~~liquid~~ water ~~relief~~ discharge events at four PWR plants. The report estimated 698 total demands on these PORVs during their ~~liquid~~ water ~~relief~~ discharge events with no failures to close. The report also summarized test data for three valve types from the Equipment Performance and Information Exchange (EPIX) database ~~for three~~

[93] NRC 2015a
[94] EPRI 2004
[95] NRC 2011

~~valve types~~maintained by the Institute of Nuclear Power Operations. The report indicates ~~2~~ two failures of PORVs to reclose during 2070 demands, but does not specify ~~liquid~~ water or steam service for the EPIX test information. With respect to PSVs, the report indicates ~~2~~ two failures out of ~~4~~ four total demands following plant scrams, but does not indicate ~~liquid~~ water or steam service. Following a request by the Panel, NRC staff from the Office of Nuclear Regulatory Research provided Licensee Event Report information indicating that the ~~2~~ two PSV failures involved incomplete reseating of the valves with leakage of 25 and 200 gallons per minute, respectively. The report summarized EPIX test data for PSVs as no failures to reclose during 1805 demands.

Plant-Specific Actions

Diablo Canyon

~~On August 13,~~In 1996, the licensee ~~of~~ for ~~the~~ Diablo Canyon Power Plant (Diablo Canyon) ~~nuclear power plant~~ submitted a report ~~under~~ of its evaluation under Title 10 of the *Code of Federal Regulations* (10 CFR)~~10 CFR~~, Section 50.59, "Changes, tests and experiments," of ~~related to~~ the potential for an IOECCS event.[96] ~~(ADAMS Microfiche 89419:294-322)~~ The submittal included NSAL-93-013 and its ~~S~~supplement 1 as enclosures. The licensee indicated that the PSVs had not been initially qualified for water discharge, but were subsequently qualified to discharge water for a brief period. The licensee indicated that WCAP-11677 was applicable and demonstrated that the PSVs were operable.

~~On July 2,~~In 2004, the NRC ~~granted~~ issued a license amendment ~~request (LAR)~~ for Diablo Canyon that allowed credit for actuation of the PORVs in response to inadvertent safety injection (SI) actuation, to avoid challenges to the PSVs.[97] ~~(ADAMS Accession No. ML041950300) In support of that LAR, the licensee responded on November 21, 2003, to requests for additional information (RAIs)~~To support the NRC staff's review, the licensee submitted additional information related to the capability of the PORVs to function adequately under conditions predicted for design-basis transients and accidents.[98] ~~(ADAMS Accession No. ML033360735)~~ In response to a question ~~n RAI~~ regarding the design adequacy of the PORVs if the pressurizer becomes water solid, the licensee referenced a January 1986 NRC letter that had ~~had stated that the NRC had issued a letter dated January 26, 1986, "Safety and Relief Valve Testing, NUREG-0737 Item II.D.1," that provided an SER that~~ accepted the adequacy of the PORV and block valve design and confirmatory testing for a range of fluid conditions (full pressure steam, steam to water transition, and subcooled water fluid).[99]

Salem

~~On June 4,~~In 1997, the NRC ~~granted a~~issued a license amendment revising the technical specification (TS) ~~revision~~ for ~~the~~ Salem ~~nuclear power plant~~Nuclear Generating Station, Units 1 and 2 (Salem) to ensure that the automatic capability of the PORVs to relieve pressure ~~is~~ would be maintained.[100] ~~(ADAMS Accession No. ML011720397)~~ In response to NSAL-93-013, the licensee determined that an inadvertent SI actuation at power could cause the pressurizer to become water solid. The ~~and~~ PSVs ~~lifting~~ would lift and discharge~~with~~ water ~~discharge~~ if the

[96] PG&E 1996
[97] NRC 2004a
[98] PG&E 2003
[99] NRC 1986a
[100] NRC 1997

automatic operation of the PORVs ~~is~~ were not made available for reactor coolant system (RCS) depressurization early in the transient. In that the Salem PSVs were not designed to relieve water, it was noted that water discharge ~~has the potential to~~could cause the PSVs to fail in the open position.

~~In the course of the review of the licensee's application~~During the review, the NRC staff noted that the PORVs were not designed to "safety related" standards and, thus, could not be credited for automatic mitigation of ~~the~~ an inadvertent SI actuation at power ~~incident when the PORV is operating in the automatic mode~~. In response, the licensee proposed an upgrade of the PORVs to eliminate the possibility that a single active failure of a PORV component could prevent the mitigation of ~~the~~ an inadvertent SI actuation at power ~~incident~~. As discussed in the ~~SER~~NRC staff's safety evaluation (SE), the licensee implemented modifications to the PORV circuitry to qualify the upgraded circuitry as safety-related.

Regarding PORV performance, the licensee evaluated the PORV air accumulators ~~for~~ and determined that they had sufficient capacity for the inadvertent SI event. The licensee also reported that endurance tests had been performed with five different trims (with different trim materials) on one PORV at Wyle Laboratories to demonstrate that (1) after 2000 consecutive operations, there were no packing leaks ~~n~~or packing gland adjustments required; (2) there was no diaphragm failure; and (3) the solenoid valve withstood 10,000 operations without any loss of function. Based on this information, the NRC staff concluded that the PORV performance was acceptable ~~regarding the mitigation of~~to mitigate an inadvertent SI event.

Millstone~~, Unit~~ 3

~~On June 5,~~In 1998, the NRC ~~granted~~ issued a license amendment for Millstone Nuclear Power Station, Unit 3 (Millstone 3) ~~for a~~that revised the TS ~~revision~~ to ensure that the capability of the PORVs to relieve pressure ~~is~~ would be maintained.[101] ~~(ADAMS Accession No. ML011800207)~~ The revised TS Bases stated that the PORVs and their associated piping ~~have~~ had been demonstrated to be "qualified" for water discharge. The PORVs would prevent water discharge from the PSVs, for which qualification for water discharge ~~has~~ had not been demonstrated. The TS Bases also stated that the prime importance for the capability to close the block valve is to isolate a stuck-open PORV. In the SE~~R~~, the NRC staff referenced a December 1997 Licensee Event Report that~~stated that the licensee~~ notified the NRC of the issue ~~of~~ of potential failure of PSVs following~~potential~~ water discharge ~~through the PSVs that could lead to valve failure in LER 97-063-00 on December 31, 1997~~.[102]

~~To provide added assurance that the PSVs will not be damaged due to water discharge during an ISI event~~As part of this license amendment, the licensee upgraded the PORV circuitry, added additional PORV surveillance requirements, qualified the PORVs and associated piping for water discharge, and ~~made~~ revised emergency procedures~~ changes~~ to allow plant operators additional time to terminate the event. With respect to the PORV circuitry, the NRC staff concluded that the PORV circuitry modifications qualified the PORV control circuitry as safety-related. With respect to PORV performance, the licensee reanalyzed the inadvertent SI event with the LOFTRAN computer code to ~~demonstrate that the PORVs were qualified for water discharge for approximately 1 hour~~determine the time available for operator action to make a PORV available and provide the mass and energy releases needed to qualify the PORVs and associated piping for water discharge. The licensee referenced EPRI testing ~~documented in NP-~~

[101] NRC 1998
[102] Northeast 1997

~~2670-LD, Volume 11,~~ that was said to generically resolve ~~post TMI-2 issues~~TMI Action Plan Items associated with PORVs and safety valve qualification for water and steam ~~relief~~discharge, ~~with~~ specifically the results from four tests of a Garrett PORV (such as used at Millstone~~, Unit~~ 3) for water discharge.[103] The licensee determined that the PORVs and associated piping are qualified for 1 hour of water discharge for an IOECCS event. The licensee also stated that the PORV manufacturer performed numerous cycle tests to verify the performance of the valve design, and also verified that valve seat leakage was acceptable. The licensee stated that the PORV block valves had been evaluated for water discharge in accordance with the program established in response to Generic Letter (GL) 89-10~~, "Safety-Related Motor-Operated Valve Testing and Surveillance."~~.[104] The NRC staff found the licensee information regarding the qualification of the PORVs for water discharge during the inadvertent SI event to be acceptable.

Callaway

~~On September 25,~~In 2000, the NRC ~~granted~~ issued a license amendment for ~~the~~ Callaway Plant, Unit 1 (Callaway)~~nuclear power plant to~~ that revised the TS to change the PSV lift setting range.[105] The changes also credited automatic actuation of at least one PORV during an IOECCS event t~~(ADAMS Accession No. ML003753326)~~ ~~T~~o prevent water ~~passing~~ discharge through the PSVs~~.~~; to enable this credit~~during an IOECCS event~~, the licensee modified and upgraded the PORV circuitry to full ~~Class~~ Class 1E ~~to take credit for automatic action of at least one PORV during the event. These actions would prevent water discharge through the PSVs.~~ In its ~~TS~~ license ~~revision~~ amendment request ~~dated May 25, 2000,~~[106] the licensee had stated that the design function of the valves was not being changed and the conclusions documented in the NRC staff's previous evaluation of ~~SER of~~ Callaway's response to ~~NUREG-0737~~TMI Action Plan Item II.D.1[107] ~~(dated September 10, 1987) are~~were also unchanged. As a result, the licensee stated that the PORVs and associated discharge piping can accommodate water discharge.

Byron and Braidwood

~~On May 29,~~In 1998, the licensee for Byron and Braidwood ~~proposed~~ requested an amendment to its TS to take credit for ~~the~~ automatic operation of the PORVs to ~~provide mitigation for~~mitigate an IOECCS event.[108] In the amendment request, the licensee stated that the PSVs ~~have~~ had not been qualified to reseat after passing subcooled liquid. The licensee stated that the PORVs at Byron and Braidwood are safety-related components with safety-related actuators and accumulator tanks, with PORV control circuits classified as safety-related. The licensee noted that some portions of the PORV circuitry are nonsafety-related, with improvements implemented in response to GL 90-06~~, Resolution of Generic Issue 70, "Power-Operated Relief Valve and Block Valve Reliability" and Generic Issue 94, "Additional Low-Temperature Over-Pressure Protection for Light-Water Reactors" Pursuant to 10 CFR 50.54(f)."~~.[109] The licensee stated that the PORV block valves are within the scope of the GL 89-10 program.

In ~~a letter dated May 13, 1999,~~1999, the NRC staff ~~provided an RAI~~requested additional information ~~regarding the reliance on the PORVs that documented the basis for its~~related to

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[103] EPRI 1982a (Volume 11)
[104] NRC 1989
[105] NRC 2000
[106] Union Electric 2000
[107] NRC 1987b
[108] ComEd 1998
[109] NRC 1990b

concerns that the PORV circuitry did not meet the single failure criterion.[110] ~~In response to these concerns, t~~The licensee reevaluated its approach and withdrew its TS amendment request ~~in a letter dated July 16, 1999~~.[111] No further action regarding this amendment request ~~has~~ was ~~been~~ identified by the Panel. However, in a public meeting during the review of the NRR Appeal,[112] the licensee stated ~~during a public meeting with the NRC staff on March 7, 2016, to discuss this backfit issue~~ that the PORVs and their block valves at Byron and Braidwood are safety-related with the exception of one circuitry aspect of the PORV.[113] ~~(see March 7, 2016, transcript, pages 36-40).~~

~~On July 5, 2000~~In 2001, the NRC issued a license amendment for~~licensee for Byron and Braidwood submitted a request for a power uprate for~~ Byron and Braidwood to increase the maximum thermal power for each unit from 3411 megawatts thermal (MWt) to 3586.6 MWt (commonly referred as a stretch power uprate).[114] ~~In RAIs~~During its review, the NRC staff requested that the licensee address water solid conditions in the pressurizer, because it had generally not accepted a solid pressurizer for an IOECCS event to order ~~to avoid~~given the potential for all three PSVs to be stuck open due to liquid relief through these safety valves. In ~~its letter dated November 27, 2000~~response, the licensee stated that Section 15.5.1, "Inadvertent Operation of Emergency Core Cooling System During Power Operation," of the UFSAR had been revised to credit the PSVs to pass water.[115] The licensee discussed the EPRI testing program in response ~~to NUREG-0737~~TMI Action Plan Item II.D.1, with the results summarized in EPRI NP-2628-SR.[116] The licensee referenced previous NRC approvals related to TMI Action Plan Item II.D.1~~the NRC letters from L. Olshan to H. Bliss, dated August 18, 1988, and S. Sands to T. Kovach, dated May 21, 1990, transmitting the TERs with the results of the NRC's review of the Byron and Braidwood response to TMI Action Plan Item II.D.1, respectively.~~.[117]

~~On January 31, 2001, the licensee for Byron and Braidwood provided a response to an RAI supplement from t~~The NRC staff ~~requesting~~ made a further request regarding the temperature of water ~~to be passed by the pressurizer safeties~~that would be discharged by the PSVs and the length of time that the ~~safeties~~ PSVs would be~~are~~ expected to ~~pass~~ discharge water. The NRC staff also asked the licensee to discuss ~~what~~ which EPRI tests are applicable to the Byron and Braidwood condition. In response, the licensee stated that the PSVs would close after ~~passing~~ discharging water, although they may not be leaktight.[118] The licensee stated that the leakage from up to three leaking PSVs is bounded by one fully open PSV. The licensee indicated that the EPRI testing of the Crosby safety valves in EPRI NP-2770-LD, Volumes 1 ~~and~~ and 6,[119] are applicable. The licensee indicated that valve chatter occurred during the tests with damage to the internals, but that the safety valve closed in response to system depressurization. The licensee stated that the Byron and ~~/~~Braidwood pressurizer water temperature of 590 °F is higher than the EPRI tests (530 °F). The licensee stated that the assumed length of the event is 20 minutes from initial SI signal to when the system pressure is restored below PSV lift setpoint.

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[110] NRC 1999
[111] ComEd 1999
[112] Exelon 2015
[113] NRC 2016a
[114] NRC 2001b
[115] ComEd 2000b
[116] EPRI 1982b
[117] NRC 1998c and NRC 1990a
[118] Exelon 2001
[119] EPRI 1982c and EPRI 1983

In Section 3.2 of the SE accompanying the license amendment~~NRC SER dated May 4, 2001, granting the Byron/Braidwood power uprate in Section 3.2, "Non-LOCA [loss-of-coolant accident] Transient Analysis,"~~, the NRC staff discussed its review of the performance of the PORVs and PSVs to discharge liquid water for approximately 20 minutes. ~~(ADAMS Accession No. ML033040016)~~ The NRC staff discussed the EPRI testing program, with the conclusion that the ~~safety valve closed~~PSV would close in response to system depressurization. The NRC staff reviewed the licensee's evaluation of the performance of the PSVs for liquid water conditions. The NRC staff found that the EPRI tests adequately demonstrated the performance of the valves for the expected water temperature conditions, and that there ~~is~~ was reasonable assurance that the valves ~~will~~ would adequately reseat following the spurious SI event. The NRC staff determined that ~~a review of the~~ EPRI test data indicated~~s~~ that the PSVs ~~may~~ might chatter for the expected fluid inlet temperature, but that the resulting PSV seat leakage following the water discharge would be less than the discharge from one stuck-open PSV. Therefore, the NRC staff found the licensee's crediting of the PSVs to discharge liquid water during the spurious SI event to be acceptable. This portion of the SE~~NRC SER~~ was based on input provided by~~the specific review of PSV performance for the Byron and Braidwood power uprate request described in a memorandum dated March 15, 2001, from~~ the ~~NRR~~ Office of Nuclear Reactor Regulation (NRR) Reactor Systems Branch, with technical input from the responsible staff member for safety valves in the NRR Division of Engineering ~~(ADAMS Accession No. ML010740316)~~.[120]

As noted by the licensee, Section 15.5.1 of the Byron and ~~/~~Braidwood UFSAR at the time of the stretch power uprate ~~(Revision 9, dated December 2002) in Chapter 15.5.1~~ includes PSV water discharge~~,~~ and references ~~the INEL 1988 report and L. Olshan August 1988 SER~~the TMI Action Plan Item II.D.1 approvals.[121] The current UFSAR Revision 15 ~~(dated December 2014)~~ concludes that the IOECCS event does not progress into a stuck-open PSV LOCA event.[122] The UFSAR states that all three PSVs may lift but will reclose, and that the leakage is bounded by one fully open valve with the consequences bounded by the IOPSRV event. The UFSAR also specifies that if ~~a safety injection~~SI results in discharge of coolant through the pressurizer valves, the operators will bring the plant to cold shutdown to inspect the valves.

~~On August 26, 2004~~In 2004, the NRC issued a license amendment for Byron and Braidwood granting an adjustment to the PSV setpoints.[123] ~~(ADAMS Accession No. ML042250531) In an RAI~~As documented in the SE, the NRC staff requested during its review that the licensee perform a quantitative analysis regarding PSV water cycles and ~~relief/~~discharge water temperature. ~~As f~~For the loss of ac power (LOAC) with reactor coolant pump (RCP) seal injection event, the licensee's analysis indicated that continued injection of water into the RCS through the RCP seals would result in a water-solid pressurizer and water discharge through the PSVs. The proposed PSV setpoint tolerance assuming negative tolerance would result in a lower PSV lift setpoint. With the lower setpoint, the PSV would open earlier, and a larger number of PSV water cycles with a lower water discharge temperature could result during the transient. The licensee performed an analysis of the LOAC with RCP seal injection event, and determined the revised PSV setpoint would result in an increase of about one PSV water cycle and a reduction in the water discharge temperature of about 0.5 °F. A comparison of the reanalysis showed that the spurious SI event remained the limiting event since it resulted in a

[120] NRC 2001a
[121] Exelon 2002
[122] Exelon 2014
[123] NRC 2004b

greater increase in the number of PSV water cycles (two cycles vs. one cycle) and a greater decrease in the PSV discharge water temperature (3.0 °F vs. 0.5 °F) than that calculated for the LOAC with RCP seal injection event. The water discharge temperature in the analysis of record for the spurious SI event was 590 °F. The lowest discharge water temperature for the spurious SI event with the revised PSV setpoint ~~is~~ was 587 °F. The NRC staff found that the calculated water discharge temperature (587 °F) ~~is~~ was significantly higher than the discharge water temperature of 530 °F that was used to support operability of the PSVs as discussed in the analysis of record. As a result, the NRC staff concluded that the ~~re~~analysis ~~is~~ was acceptable to assure that the PSVs will remain operable following a spurious SI event.

~~On February 7,~~ In 2014, the NRC issued a license amendment for Byron and Braidwood granting a m~~M~~easurement u~~U~~ncertainty r~~R~~ecapture (MUR) power uprate.[124] The NRC staff determined that the IOECCS event was outside of the scope of the MUR power uprate, because the licensee did not ~~modify~~ propose to modify the Chapter 15 analyses related to PSV and PORV water discharge.

With respect to inservice testing (IST) activities, the Byron IST ~~P~~program[125] ~~(dated July 21, 2016)~~ references the ASME *Code for Operation and Maintenance of Nuclear Power Plants* (OM Code), 2004 Edition through 2006 Addenda; and the Braidwood IST p~~P~~rogram[126] ~~(dated July 27, 2009)~~ references the ASME OM Code, 2001 Edition through 2003 Addenda. The Byron IST Program specifies the following testing and intervals for the PORVs, PORV block valves, and PSVs:

- **PORV**: ~~-~~ fail safe test closed (cold shutdown interval), ~~;~~ stroke-time exercise open and closed (cold shutdown interval), ~~;~~ and position indication test (2 year interval)~~.~~
- **PORV Block Valve**: ~~-~~ exercise open and closed (2 year interval); position indication test (Joint Owners Group (JOG) Program interval); and open and closed test in accordance with ASME OM Code Case OMN-1, "Alternative Rules for Preservice and Inservice Testing of Active Electric Motor Operated Valve Assemblies in Light-Water Reactor Power Plants" (JOG Program interval)~~.~~
- **PSV**: ~~– **Relief Valve Test (5 year interval); and**~~ position indication test (2 year interval) and relief valve test (5 year interval), referencing~~. The Byron IST Program references~~ ASME OM Code, Appendix I, "Inservice Testing of Pressure Relief Devices in Light-Water Reactor Nuclear Power Plants~~." to the ASME OM Code for Relief Valve Testing.~~"

The Braidwood IST Program specifies the following testing and intervals for the PORVs, PORV block valves, and PSVs:

- **PORV**: ~~-~~ fail safe test closed (refueling outage interval), ~~;~~ stroke-time exercise open and closed (refueling outage interval), ~~;~~ and position indication test (2 year interval).
- **PORV Block Valve**: ~~-~~ exercise open and closed (quarterly interval)~~;~~ and position indication test (2 year interval)~~.~~

[124] NRC 2014a
[125] Exelon 2016b
[126] Exelon 2009

- **PSV**: ~~–~~ ~~Relief Valve Test (5 year interval); and~~ position indication test (2 year interval), and relief valve test (5 year interval), referencing ASME OM Code, Appendix I.~~ The Braidwood IST Program references Appendix I to the ASME OM Code for Relief Valve Testing.~~

Shearon Harris

~~On~~ In ~~October 12,~~ 2001, the NRC ~~granted~~ issued a license amendment to ~~the~~ Shearon Harris N~~n~~ucle~~e~~ar P~~p~~ower P~~p~~lant, Unit 1 (Shearon Harris) for steam generator replacement and a power uprate to a maximum power level of 2900 MWt (approximately 4.5 percent).[127] In addressing the licensee's evaluation of ~~SRP~~ Standard Review Plan (SRP) Section 15.5.1, the NRC staff found that the analysis showed that the calculated inlet pressures and temperatures required for the PORVs and ~~SRVs~~ safety relief valves (SRVs)[128] to operate in a water environment ~~are~~ were within the valve operable ranges, and thus ensured that the PORV and SRV ~~are~~ would be operable during the transient. The valve operable ranges were previously determined by the licensee to support operability of the PORV and SRV during the discharge of subcooled water in accordance with the TMI Action Plan Item II.D.1 requirements. Based on the analysis meeting the acceptance criteria of SRP Section 15.5.1 with respect to the RCS pressure limit and departure-from-nucleate-boiling limit, the NRC staff concluded that the analysis was acceptable.

Beaver Valley

~~On July 19,~~In 2006, the NRC ~~granted~~ issued a license amendment authorizing an EPU ~~to~~ for Beaver Valley Power Station, Units 1 and 2 (~~BVPS-1 and 2~~Beaver Valley), ~~for~~ an approximate 8-percent increase in thermal power to 2,900 MWt.[129] In ~~its SER (ADAMS No. ML061720376)~~the SE accompanying the amendment, the NRC staff ~~stated that a specific issue which was reviewed related to~~described its review of the capability of the PSVs to discharge liquid and adequately reseat for a spurious SI actuation. The ~~specific issue which the NRC staff evaluated in this regard was~~NRC staff specifically evaluated whether the PSVs could reasonably be expected to reseat ~~in order~~ to prevent the spurious SI actuation (an ANS Condition II event) from causing a stuck-open PSV (an ANS Condition III event). This issue was said to be further discussed in RIS 2005-29. While the PSVs ~~are~~ for Beaver Valley were qualified to discharge steam, if the valves discharged ~~liquid~~ water ~~having a temperature low enough~~with sufficient subcooling, ~~they may~~the NRC staff was concerned that they might not reseat properly.

Based the licensee's analysis, during a spurious SI event, the PSVs would be required to discharge steam followed by high temperature ~~liquid~~ water after the pressurizer filled~~s~~. The licensee provided plots of the pressurizer water temperatures for this event ~~which~~ that indicated that the minimum temperature of the discharged liquid for ~~both BVPS-1 and 2 is~~Beaver Valley was approximately 620 °F. To evaluate the capability of the valves to discharge and reseat, the NRC staff reviewed the available data from the full-flow tests performed during the EPRI test program in 1981 for the specific PSV models representative of those installed at ~~BVPS-1 and 2~~Beaver Valley. The licensee also used the methodology contained in WCAP-11677~~,~~ and determined that the minimum acceptable liquid temperature for which the PSVs ~~are~~ were

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[127] NRC 2001d

[128] This term is used in the Shearon Harris SE; the Panel considers the term SRV to be equivalent to PSV for this facility.

[129] NRC 2006

expected to successfully discharge and reseat ~~is~~ was less than the minimum expected temperature for the spurious SI event for ~~BVPS-1 and 2~~Beaver Valley.

The NRC staff agreed that both the minimum expected water discharge temperature and the minimum acceptable ~~liquid~~ water temperature had been conservatively calculated. Therefore, the NRC staff determined that, for purposes of preventing the occurrence of a more serious ANS Condition III event, there ~~is~~ was reasonable assurance that the PSVs would ~~adequately~~ discharge water and reseat adequately following a spurious SI actuation. A consideration of the NRC staff in making this finding was that, in the unlikely event of a stuck-open PSV, the ECCS ~~is~~ was fully capable of mitigating the resulting LOCA.

Turkey Point

~~On June 15,~~In 2012, the NRC ~~granted~~ issued a license amendment authorizing an EPU for Turkey Point Nuclear Generating, Units 3 and 4 (Turkey Point), increasing ~~that increased~~ the thermal power level of each unit approximately 15 percent to 2644 MWt.[130]

In the SE accompanying the amendment~~R (ADAMS Accession No. ML11293A359)~~, the NRC staff indicated that ECCS actuation ~~is~~ was not a possible initiator of inadvertent increase in reactor coolant inventory because the high head SI pumps have a shut-off head below the normal RCS operating pressure. The NRC staff stated that a CVCS malfunction that increases RCS inventory was evaluated for the effects of adding water inventory to the RCS. If the pressurizer filled~~s~~ and caused~~s~~ water to be relieved through the PORVs or ~~safety valves~~PSVs, then these valves could stick open and create a small break LOCA. The NRC staff stated that this would violate the acceptance criterion that prohibits the escalation of an anticipated operational occurrence (AOO) into a more serious event. Satisfaction of this acceptance criterion ~~is~~ was demonstrated by showing that sufficient time ~~exists~~ would exist for the operator to recognize the situation and end the charging flow before the pressurizer ~~can~~ could fill. The NRC staff concluded that the licensee's analyses of IOECCS and CVCS events adequately accounted for operation of the plant at the proposed power level.

Regarding an inadvertent opening of a ~~pressurizer relief valve~~PORV, the licensee initially proposed that the consequences of this event ~~are~~ were bounded by the small break LOCA. The NRC staff did not accept this proposed disposition. If action ~~is~~ were not taken to secure the open valve by either closing the PORV or its block valve, the NRC staff stated that this event could escalate to a small-break LOCA, which ~~is~~ would be contrary to the non-escalation ~~criterion~~position. When the pressurizer ~~becomes water solid~~filled, water ~~begins~~ would begin to flow through the open PORV. If the PORV ~~is~~ were not qualified for water discharge, the NRC staff stated that it ~~is~~ was likely the PORV ~~will~~ would not close upon demand. In this way, the NRC staff stated that the inadvertent opening of a PORV, an AOO, would become~~s~~ a small break-LOCA at the top of the pressurizer, an ANS Condition III event. The NRC staff requested that the licensee address the inadvertent opening of the PORV with respect to the third criterion for an ANS Condition II event.

The licensee provided an analysis~~,~~ performed largely in accordance with NRC-approved, Westinghouse analytic methodology using the RETRAN computer code; however, this analysis was performed assuming that the PORV opened instead of the PSV. The NRC staff stated that

[130] NRC 2012a

assuming the opening of the PORV is acceptable, because the PSV is differently qualified, and reseats mechanically. An additional independent fault would be required to cause the ~~safety valve~~ PSV to fail to close. The analysis indicated that the pressurizer would fill within about 240 seconds. The licensee stated that there ~~are~~ were multiple alarms to indicate the opening of a PORV. The licensee stated that a prompt operator action ~~is~~ would be ~~required~~ needed to close the PORV and, if the PORV does not close, the operator ~~is to~~would be directed to close the block valve. Because the necessary actions ~~are~~ would be prompt and simple, the NRC staff agreed that there ~~is~~ would be sufficient time to secure the inadvertently open PORV without filling the pressurizer.

St. Lucie

~~On September 24,~~In 2012, the NRC ~~granted~~ issued a license amendment authorizing an EPU for St. Lucie Plant, Unit 2 (St. Lucie, Unit 2) that increased the authorized thermal power level about 12 percent to ~~3020~~ 3020 MWt.

~~-~~Regarding an IOECCS event, the high pressure SI pumps ~~are~~ would be incapable during power operations of delivering flow to the RCS because the pumps' shut-off head ~~is~~ would be less than the normal RCS operating pressure of 2250 pounds per square inch absolute. Therefore, the licensee determined that the inadvertent operation of the ECCS at power event ~~is~~ was not a credible event ~~and was not analyzed by the licensee~~and did not analyze it for the proposed EPU. The NRC staff found that the licensee's position for not analyzing the IOECCS event to be acceptable.

Regarding a CVCS malfunction, ~~this event increases RCS inventory as an AOO that is~~the licensee evaluated it as an AOO ~~evaluated for~~ for the effects of adding water inventory to the RCS. The NRC staff reviewed the licensee's analyses of the CVCS malfunction event and concluded that the licensee's analyses adequately accounted for operation of the plant at the proposed power level and were performed using acceptable analytical models. The NRC staff determined that the licensee's analysis demonstrated that the pressurizer did not become water solid, assuring no water was discharged through the PSVs.

Regarding an IOPORV event, the NRC staff stated that, when viewed from the mass addition perspective, this event ~~can~~ could be evaluated in two phases: (1) an inadvertent opening of a pressurizer relief valve, followed by (2) an inadvertent ECCS actuation. In the first phase, the NRC staff stated that this event could be mitigated by closing the open ~~pressurizer relief valve~~PORV or its block valve. If the PORV or its block valve was not closed, the NRC staff stated that the IOPORV event would enter the second phase with actuation of the ECCS. Based on its review, the NRC staff determined that the pressurizer overfill analysis, available alarming system, and procedures, in combination with simulator exercise results, ~~had~~ provided reasonable assurance that the pressurizer would not be expected to fill to a water solid condition that could prevent the PORV or PSV from closing after they were open. The NRC staff therefore concluded ~~, and thus, supported t~~that the event would not generate a more serious plant conditions, meeting the non-escalation criterion. The NRC staff stated that it reviewed the licensee's analyses of the inadvertent opening of a pressurizer ~~pressure relief valve~~PORV event, and concluded that the licensee's analyses adequately accounted for operation of the plant at the proposed power level and were performed using acceptable analytical models.

The NRC staff concluded that the licensee demonstrated that ~~the~~ all AOO acceptance criteria ~~are~~ were satisfactorily met.

North Anna

North Anna Power Station, Units 1 and 2 (North Anna) ~~In~~ UFSAR ~~(Revision 51, dated September 30, 2015)~~ Section 15.2.14, "Spurious Operation of the Safety Injection System at Power," ~~the licensee for North Anna Units 1 and 2 discusses~~describes ~~the~~ plant response to an inadvertent SI event. In particular, UFSAR Section 15.2.14.2.3, "Event Propagation," states the following:

> Safety valve (Reference 18) and PORV (Reference 19) testing has revealed no instances of failure of the valves to reseat following water relief. Resulting leakage is within the capacity of the normal makeup system and is therefore not considered to be a small break loss of reactor coolant event. Therefore, the complete filling of the pressurizer and/or water relief via a safety valve as a result of a spurious safety injection does not constitute a failure to meet the event propagation acceptance criterion. Although primary credit for preventing the propagation of the event to a small break loss of reactor coolant event is the reseating of the PORVs and safety valves, it is noted that the PORVs (which open prior to the safety valves and, if open, preclude safety valve actuation for this event) are provided with block valves which the operator will close in the event of excessive PORV leakage.

North Anna UFSAR Section 15.2.14.3, "Conclusions," ~~states~~ indicates that the complete filling of the pressurizer and~~/or~~ water discharge via a ~~safety valve~~PSV as a result of a spurious ~~safety injection~~SI does not constitute a failure to meet the ~~event propagation acceptance criterion~~non-escalation position. Furthermore, ~~In~~ UFSAR Section 15.2, "References," lists EPRI NP-2770-LD and EPRI NP-2670-LD.~~Reference 18 as EPRI NP-2770-LD, Volumes 3 and 4, "EPRI/CE PWR Safety Valve Test Reports for Dresser Safety Valve Models 31739A and 31709NA," February and March 1983; and Reference 19 as EPRI NP-2670-LD, Volume 6, "EPRI/Wyle Power-Operated Relief Valve Phase III Test Report, October 1982.~~

Conclusion

In conclusion, the reliance by the licensee for Byron and Braidwood on the acceptable performance of the PSVs and PORVs ~~for liquid service~~following water discharge in response to abnormal events is not inconsistent with similar approaches by some other nuclear power plant licensees. In general, the review of activities by various nuclear power plant licensees related to PSV and PORV performance revealed reliance on EPRI, Wyle, and valve vendor testing to provide support for the performance of these valves under various service conditions. Specific certification for flow capacity of these valves for ~~liquid service~~water discharge in accordance with the ASME BPV Code and National Board was not identified in the review of various justifications prepared by nuclear power plant licensees.

In evaluating the historical documents for Byron and Braidwood, the Panel found it challenging to determine specifically how the licensee resolved the concern raised in NSAL-93-013 in its analyses and plant operations. While the record does support a compliance backfit in this case, if (as recommended by the Panel) the NRC staff undertakes a generic review of licensees' treatment of the potential for pressurizer valve damage following water discharge, it may be appropriate to consider what actions have been taken, how operating experience with water discharge has been considered, and how analysis assumptions are considered in operational practices (including inservice testing) at each plant.

APPENDIX D: REFERENCES

1. **AEC 1970:** Atomic Energy Commission (AEC), "Backfitting of Production and Utilization Facilities; Construction Permits and Operating Licenses," Title 10 of the *Code of Federal Regulations* (10 CFR), Section 50.109, published March 31, 1970 (as amended). Volume 35 of the *Federal Register* (FR), page 5317.

2. **AEC 1971:** AEC, "General Design Criteria for Nuclear Power Plants," 10 CFR Part 50, Appendix A, first published February 20, 1971 (as amended). 36 FR 3256.

3. **ANS 1973:** American Nuclear Society (ANS), ANS-51.1/N18.2-1973, "Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants," dated August 6, 1973. [Publicly available from ANS for a fee.]

4. **ANS 1981:** ANS and American National Standards Institute (ANSI), ANSI/ANS 58.9, "Single Failure Criteria for Light Water Reactor Safety-Related Fluid Systems," issued February 17, 1981 (updated in 2002 and reaffirmed in 2015). [Publicly available from ANS for a fee.]

5. **ANS 2016:** ANS, "American Nuclear Society Standards Committee Glossary of Definitions and Terminology," dated May 19, 2016. Available from ANS at http://cdn.ans.org/standards/resources/toolkit/docs/glossary-of-definitions.pdf.

6. **ComEd 1982:** Commonwealth Edison (ComEd), letter from T. R. Tramm to Harold R. Denton, U.S. Nuclear Regulatory Commission (NRC), "Byron Station Units 1 and 2; Braidwood Station Units 1 and 2; Pressurizer Safety and Relief Valves," dated October 26, 1982. [This document is not in ADAMS, but is available through the NRC Public Document Room using Accession No. 8211020633 and microfiche location 15886:171 – 15886:231.]

7. **ComEd 1998:** Commonwealth Edison Company, letter from K.L. Grasser to U.S. NRC, "Change to Credit Automatic PORV Operation for Mitigation of Inadvertent Safety Injection at Power Accident," dated May 29, 1998. [This document is not in ADAMS, but is available through the NRC Public Document Room using Accession No. 9806020180 and microfiche location A3765:275-301.]

8. **ComEd 1999:** Commonwealth Edison Company, letter from R.M. Krich to U.S. NRC, "Response to Request for Additional Information Regarding License Amendment Request to Credit Automatic Power-Operated Relief Valve (PORV) Operation for Mitigation of Inadvertent Safety Injection at Power Accident and Withdrawal of License Amendment Request," dated July 16, 1999. [This document is not in ADAMS, but is available through the NRC Public Document Room using Accession No. 9907200183 and microfiche location A8671:349-354.]

9. **ComEd 2000a:** Commonwealth Edison Company, letter from R.M. Krich to U.S. NRC, "Request for a License Amendment to Permit Uprated Power Operations at Byron and Braidwood Stations," dated July 5, 2000. ADAMS Accession No. ML003730608.

10. **ComEd 2000b:** Commonwealth Edison Company, letter from R.M. Krich 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," dated November 27, 2000. ADAMS Accession No. ML003772461.

11. **Consumers 1982:** Consumers Energy Company, letter from D. P. Hoffman to Harold R. Denton, U.S. NRC, "Safety and Relief Valve Test Report for the EPRI PWR Safety and Relief Valve Test Program," dated April 1, 1982. [This document is not in ADAMS, but is available through the NRC Public Document Room using Accession No. 8207160337 and microfiche location 13866:001 – 13869:106.]

12. **EPRI 1982a:** EPRI, NP-2670-LD, "EPRI/Wyle Power-Operated Relief Valve Phase III Test Report," dated October 1982. [Publicly available from EPRI for a fee.]

13. **EPRI 1982b:** EPRI, NP-2628-SR, "EPRI PWR Safety and Relief Valve Test Program – Safety and Relief Valve Test Report," dated December 1982. [This document is not in ADAMS, but is available through the NRC Public Document Room using Accession No. 8407130197 and microfiche location 25588:082-262.]

14. **EPRI 1982c:** Electric Power Research Institute (EPRI), NP-2770-LD, Volume 1, "EPRI/C-E PWR Safety Valve Test Report: Summary," dated December 1982. Publicly available from EPRI.

15. **EPRI 1983:** Electric Power Research Institute (EPRI), NP-2770-LD, Volume 6, "EPRI/C-E PWR Safety Valve Test Report: Test Results for Crosby Safety Valve," dated March 1983. [Publicly available from EPRI for a fee.]

16. ~~**EPRI 1982a:** Electric Power Research Institute (EPRI), NP-2770-LD, "EPRI/C-E PWR Safety Valve Test Report," dated January 1982. [Publicly available from EPRI for a fee.]~~

16. ~~**EPRI 1982b:** EPRI, NP-2670-LD, "EPRI/Wyle Power-Operated Relief Valve Phase III Test Report," dated October 1982. [Publicly available from EPRI for a fee.]~~

16. ~~**EPRI 1982c:** EPRI, NP-2628-SR, "EPRI PWR Safety and Relief Valve Test Program – Safety and Relief Valve Test Report," dated December 1982. [This document is not in ADAMS, but is available through the NRC Public Document Room using Accession No. 8407130197 and microfiche location 25588:082-262.]~~

16. **EPRI 2004:** EPRI, TR-1011047, "Probability of Safety Valve Failure-to-Reseat Following Steam and Liquid Relief - Quantitative Expert Elicitation," dated August 2004. Publicly available from EPRI.

17. **Exelon 2001:** Exelon Generation Company, LLC, letter from R.M. Krich 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," dated January 31, 2001. ADAMS Accession No. ML010330145.

18. **Exelon 2002:** Exelon Generation Company, LLC, letter from R. M. Krich to U.S. NRC, "Byron/Braidwood Stations Updated Final Safety Analysis Report, Revision 9; Byron Station Technical Requirements Manual, Revision 32; Byron Station Technical Specifications Bases, Revision 31; Braidwood Station Technical Requirements Manual, Revision 26; Braidwood Station Technical Specifications Bases, Revision 36," dated December 16, 2002. [Non-publicly available; electronic files are in the NRC file center, and the transmittal letter is at ADAMS Accession No. ML023650683.]

19. **Exelon 2009:** Exelon Generation Company, LLC, letter from Bryan Hanson to U.S. NRC, "Braidwood Station, Units 1 and 2 – Submittal of Inservice Testing Program Plan

for the Third Ten-Year Interval," dated July 27, 2009. ADAMS Accession No. ML092110538.

20. **Exelon 2014:** Exelon Generation Company, LLC, "Byron/Braidwood Nuclear Stations Updated Final Safety Analysis Report (UFSAR)," Revision 15, dated December 2014. ADAMS Accession No. ML14363A393.

21. **Exelon 2015:** Exelon Generation Company, LLC, letter from J. Bradley Fewell to William M. Dean, U.S. NRC, "Appeal of Imposition of Backfit Regarding Compliance with Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.34(b), General Design Criteria (GDC) 15, GDC 21, GDC 29, and Licensing Basis," dated December 8, 2015. ADAMS Accession No. ML15342A112.

22. **Exelon 2016a:** Exelon Generation Company, LLC, letter from J. Bradley Fewell to Victor M. McCree, U.S. NRC, "Appeal of Imposition of Backfit Regarding Compliance with 10 CFR § 50.34(b), General Design Criteria (GDC) 15, GDC 21, GDC 29, and Licensing Basis," dated June 2, 2016. ADAMS Accession No. ML16154A254.

23. **Exelon 2016b:** Exelon Generation Company, LLC, letter from Thomas D. Chalmers to U.S. NRC, "Byron Station, Units 1 and 2, Transmittal of Inservice Testing Program for the Fourth Ten-Year Interval," dated July 21, 2016. ADAMS Accession No. ML16203A108.

24. **IAEA 2009:** International Atomic Energy Agency (IAEA), IAEA-TECDOC-1624, "Passive Safety Systems and Natural Circulation in Water Cooled Nuclear Power Plants," dated November 2009. Publicly available from IAEA.

25. **Northeast 1997:** Northeast Nuclear Energy Company, Licensee Event Report 97-063-00, "Millstone Nuclear Power Station Unit 3 – Inadequate Operator Response Time for Inadvertent Safety Injection (SI) Event," dated December 31, 1997. Text available through ADAMS Public Legacy Library using Accession No. 9802110033.

26. **NEI 2016:** Nuclear Energy Institute (NEI), letter from Anthony R. Pietrangelo to Victor M. McCree, U.S. NRC, "Nuclear Energy Institute Comments in Support of Exelon Generation Company Second-Level Backfit Appeal," dated June 16, 2016. ADAMS Accession No. ML16208A008.

27. **NRC 1976:** U.S. NRC, NUREG-0053, "Safety Evaluation Report Related to the Operation of North Anna Power Station," dated April 1976. [This document is not in ADAMS, but is available through the NRC Public Document Room using Accession No. 8001100821 and microfiche location 04043:051-278.]

28. **NRC 1977:** U.S. NRC, SECY-77-439, "Single Failure Criterion," dated August 17, 1977. ADAMS Accession No. ML060260236.

29. **NRC 1979a:** U.S. NRC, NUREG-0578, "TMI-2 Lessons Learned Task Force Status Report and Short-Term Recommendations," dated July 1979. ADAMS Accession No. ML090060030. (Transmitted to licensees via Generic Letter 79-32, "TMI-2 Lessons Learned Task Force Report – NUREG-0578," dated July 25, 1979. Publicly available on the NRC website.)

30. **NRC 1979b:** U.S. NRC, NUREG-0585, "TMI-2 Lessons Learned Task Force Final Report," dated October 1979. ADAMS Accession No. ML061430367.

31. **NRC 1980a:** U.S. NRC, NUREG-0660, "NRC Action Plan Developed as a Result of the TMI-2 Accident," dated May 1980. ADAMS Accession Nos. ML072470526 and ML072470524.

32. **NRC 1980ba:** U.S. NRC, letter from Darrell G. Eisenhut, "Post-TMI Requirements," dated October 31, 1980. Text available through ADAMS Public Legacy Library using Accession No. 8012160050.

33. **NRC 1980cb:** U.S. NRC, NUREG-0737, "Clarification of TMI Action Plan Requirements," dated November 1980. ADAMS Accession No. ML051400209.

34. **NRC 1981a:** U.S. NRC, NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition" [SRP], Section 15.0, "Introduction," Revision 2, dated July 1981. ADAMS Accession No. ML052350113.

35. **NRC 1981b:** U.S. NRC, NUREG-0800, SRP Section 15.5.1 – 15.5.2, "Inadvertent Operation of ECCS and Chemical and Volume Control System Malfunction that Increases Reactor Coolant Inventory," Revision 1, dated July 1981. ADAMS Accession No. ML052350142.

36. **NRC 1981c:** U.S. NRC, NUREG-0800, SRP Section 15.6.1, "Inadvertent Opening of a PWR Pressurizer Pressure Relief Valve or a BWR Pressure Relief Valve," Revision 1, dated July 1981. ADAMS Accession No. ML052350144.

37. **NRC 1982:** U.S. NRC, memorandum from R.L. Tedesco to R.F. Fraley, "Issuance of the Draft Safety Evaluation Report for the Watts Bar Nuclear Plant, Units 1 and 2," dated January 5, 1982. ADAMS Accession No. ML073580524 [non-public].

38. **NRC 1983:** U.S. NRC, NUREG-1002, "Safety Evaluation Report Related to the Operation of Braidwood Station, Units 1 and 2," dated November 1983. ADAMS Accession No. ML12114A272.

39. **NRC 1984:** U.S. NRC, NUREG-0876, Supplement No. 5, "Safety Evaluation Report Related to the Operation of Byron Station, Units 1 and 2," dated October 1984. ADAMS Accession No. ML091340252.

40. **NRC 1985:** U.S. NRC, "Revision of Backfitting Process for Power Reactors - Parts 2 and 50," dated September 20, 1985. 50 FR 38097.

41. **NRC 1986a:** U.S. NRC, letter from H. Schierling to J.D. Shiffer, Pacific Gas and Electric Company (PG&E), "Safety and Relief Valve Testing, NUREG-0737 Item II.D.1," dated January 27, 1986. [This document is not in ADAMS, but is available through the NRC Public Document Room using Accession No. 8602180450 and microfiche location 34630:271-299.]

42. **NRC 1986b:** U.S. NRC, NUREG-1002, Supplement No. 1, "Safety Evaluation Report Related to the Operation of Braidwood Station, Units 1 and 2," dated September 1986. [This document is not in ADAMS, but is available through the NRC Public Document

Room using Accession No. 8610220157 and microfiche location 38332:281 – 38333:087.]

43. **NRC 1987a:** U.S. NRC, NUREG-0876, Supplement No. 8, "Safety Evaluation Report Related to the Operation of Byron Station, Units 1 and 2," dated March 1987. [This document is not in ADAMS, but is available through the NRC Public Document Room using Accession No. 8704010167 and microfiche location 40321:134-156.]

44. **NRC 1987b:** U.S. NRC, letter from Thomas W. Alexion to D.F. Schnell, Union Electric Company, dated September 10, 1987. [This document is not in ADAMS, but is available through the NRC Public Document Room using Accession No. 8709170429 and microfiche location 42712:212-229.]

45. **NRC 1988a:** U.S. NRC, SECY-88-102, "Final Backfit Rule," dated April 18, 1988. [This document is not in ADAMS, but is available through the NRC Public Document Room using Accession No. 8805130269 and microfiche location 45795:299-362.]

46. **NRC 1988b:** U.S. NRC, "Revision of Backfitting Process for Power Reactors - Final Rule," dated June 6, 1988. 53 FR 20603.

47. **NRC 1988c:** U.S. NRC, letter from L. Olshan to Henry E. Bliss, Commonwealth Edison Company, "NUREG-0737, Item II.D.1, Performance Testing on Relief and Safety Valves for Byron Station, Units 1 and 2," dated August 18, 1988. ADAMS Accession No. ML003772409 [pages 161-188 of file].

48. **NRC 1988d:** U.S. NRC, NRC Manual Chapter 0514, "NRC Program for Management of Plant-Specific Backfitting of Nuclear Power Plants," dated August 26, 1988. ADAMS Accession No. ML041400111.

49. **NRC 1988e:** U.S. NRC, Bulletin 88-11, "Pressurizer Surge Line Thermal Stratification," dated December 20, 1988. Publicly available on the NRC website.

50. **NRC 1989:** U.S. NRC, Generic Letter (GL) 89-10, "Safety-Related Motor-Operated Valve Testing and Surveillance," dated June 28, 1989. ADAMS Accession No. ML031150300.

51. **NRC 1990a:** U.S. NRC, letter from S. Sands to Thomas J. Kovach, Commonwealth Edison Company, "NUREG-0737, Item II.D.1, "Performance Testing on Relief and Safety Valves for Braidwood Station, Units 1 and 2," dated May 21, 1990. ADAMS Accession No. ML003772409 [pages 189-217 of file].

52. **NRC 1990b:** U.S. NRC, GL 90-06, "Resolution of Generic Issue 70, 'Power-Operated Relief Valve and Block Valve Reliability' and Generic Issue 94, 'Additional Low-Temperature Over Pressure Protection for Light-Water Reactors,' Pursuant to 10 CFR 50.54(f)," dated June 25, 1990. ADAMS Accession No. ML031210416.

53. **NRC 1990c:** U.S. NRC, NUREG-1409, "Backfitting Guidelines," dated July 1990. ADAMS Accession No. ML032230247.

54. **NRC 1991:** U.S. NRC, letter from Anthony H. Hsia to Thomas J. Kovach, Commonwealth Edison Company, "Issuance of Amendments [for Byron and Braidwood]

(TAC Nos. M77332, M77333, M77334, M77335, M77402, M77403, M77404, and M77405)," dated November 18, 1991. ADAMS Accession No. ML020860105.

55. **NRC 1994a:** U.S. NRC, SECY-94-084, "Policy and Technical Issues Associated with the Regulatory Treatment of Non-Safety Systems in Passive Plant Designs," dated March 28, 1994. ADAMS Accession No. ML003708068.

56. **NRC 1994b:** U.S. NRC, SRM-SECY-94-084, "SECY-94-084 – Policy and Technical Issues Associated with the Regulatory Treatment of Non-Safety Systems in Passive Plant Designs and COMSECY-94-024 – Implementation of Design Certification and Light-Water Reactor Design Issues," dated June 30, 1994. ADAMS Accession No. ML003708098.

57. **NRC 1996:** U.S. NRC, NUREG-0800, SRP Section 15.5.1 – 15.5.2, "Inadvertent Operation of ECCS and Chemical and Volume Control System Malfunction that Increases Reactor Coolant Inventory," draft Revision 2, dated April 1996. ADAMS Accession No. ML052070725.

58. **NRC 1997:** U.S. NRC, letter from Leonard N. Olshan, NRC, to Leon R. Eliason, Public Service Electric & Gas Company, "Salem Nuclear Generating Station, Unit Nos. 1 and 2 (TAC Nos. M97827 and M97828)," dated June 4, 1997. ADAMS Accession No. ML011720397.

59. **NRC 1998:** U.S. NRC, letter from James W. Andersen to Martin L. Bowling, Jr., Northeast Nuclear Energy Company, "Issuance of Amendment – Millstone Nuclear Power Station, Unit No. 3 (TAC No. MA1527)," dated June 5, 1998. ADAMS Accession No. ML011800207.

60. **NRC 1999:** U.S. NRC, letter from John B. Hickman to Oliver D. Kingsley, Commonwealth Edison Company, "Request for Additional Information – Byron Station, Units 1 and 2 and Braidwood Station, Units 1 and 2 (TAC Nos. MA2043, MA2044, MA2048, and MA2049," dated May 13, 1999. [This document is not in ADAMS, but is available through the NRC Public Document Room using Accession No. 9905170241 and microfiche location A8035:313-316.]

61. **NRC 2000:** U.S. NRC, letter from Jack Donohew to Garry L. Randolph, Union Electric Company, "Callaway Plant, Unit 1 – Issuance of Amendment Re: Pressurizer Safety Valves and Power Operated Relief Valves (PORVs) (TAC No. MA9080)," dated September 26, 2000. ADAMS Accession No. ML003753326.

62. **NRC 2001a:** U.S. NRC, memorandum from Frank Akstulewicz to Anthony Mendiola, "Byron and Braidwood Stations, Units 1 and 2 – Requests for a License Amendment to Permit Uprated Power Operations (TAC Nos. MA9426, MA9427, MA9428 and MA9429)," dated March 15, 2001. ADAMS Accession No. ML010740316 [non-public].

63. **NRC 2001b:** 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.

64. **NRC 2001c:** U.S. NRC, NUREG-1800, "Standard Review Plan for Review of License Renewal Applications for Nuclear Power Plants," dated July 2001. ADAMS Accession No. ML012070413.

65. **NRC 2001d:** U.S. NRC, letter from N. Kalyanam to James Scarola, Carolina Power & Light Company, "Shearon Harris Nuclear Power Plant, Unit 1 – Issuance of Amendment Re: Steam Generator Replacement and Power Uprate (TAC Nos. MB0199 and MB0782)," dated October 12, 2001. ADAMS Accession No. ML012880381.

66. **NRC 2003:** U.S. NRC, Review Standard (RS) 001, "Review Standard for Extended Power Uprates," dated December 2003. ADAMS Accession No. ML033640024.

67. **NRC 2004a:** U.S. NRC, letter from Girija S. Shukla to Gregory M. Rueger, ~~Pacific Gas and Electric Company~~PG&E, "Diablo Canyon Power Plant, Unit Nos. 1 and 2 – Issuance of Amendment Re: Credit for Automatic Actuation of Pressurizer Power Operated Relief Valves (TAC Nos. MB6758 and MB6759)," dated July 2, 2004. ADAMS Accession No. ML041950300.

68. **NRC 2004b:** U.S. NRC, letter from George F. Dick, Jr., to Christopher M. Crane, Exelon Generation Company, LLC, "Byron Station, Units 1 and 2, and Braidwood Station, Units 1 and 2 – Issuance of Amendments, Re: Pressurizer Safety Valve Setpoints, (TAC Nos. MB9762, MB9763, MB9760, and MB9761)," dated August 26, 2004. ADAMS Accession No. ML042250531.

69. **NRC 2005a:** U.S. NRC, SECY-05-0138, "Risk-Informed and Performance-Based Alternatives to the Single-Failure Criterion," dated August 4, 2005. ADAMS Accession No. ML051950610.

70. **NRC 2005b:** U.S. NRC, Regulatory Issue Summary 2005-29, "Anticipated Transients that Could Develop into More Serious Events," dated December 14, 2005. ADAMS Accession No. ML051890212.

71. **NRC 2006:** U.S. NRC, letter from Timothy G. Colburn to James H. Lash, FirstEnergy Nuclear Operating Company, "Beaver Valley Power Station, Unit Nos. 1 and 2 (BVPS-1 and 2) – Issuance of Amendment Regarding the 8-Percent Extended Power Uprate (TAC Nos. MC4645 and MC4646)," dated July 19, 2006. ADAMS Accession No. ML061720274.

72. **NRC 2007a:** U.S. NRC, NUREG-0800, SRP Section 15.0, "Introduction - Transient and Accident Analyses," Revision 3, dated March 2007. ADAMS Accession No. ML070710376.

73. **NRC 2007b:** U.S. NRC, NUREG-0800, SRP Section 15.~~1~~5.1 – 15.5~~1~~.2, "Inadvertent Operation of ECCS and Chemical and Volume Control System Malfunction that Increases Reactor Coolant Inventory," Revision 2, dated March 2007. ADAMS Accession No. ML070820081.

74. **NRC 2007c:** U.S. NRC, NUREG-0800, SRP Section 15.6.1, "Inadvertent Opening of a PWR Pressurizer Pressure Relief Valve or a BWR Pressure Relief Valve," Revision 2, dated March 2007. ADAMS Accession No. ML070820094.

75. **NRC 2011:** U.S. NRC, NUREG/CR-7037, "Industry Performance of Relief Valves at U.S. Commercial Nuclear Power Plants through 2007," dated March 2011. ADAMS Accession No. ML110980205.

76. **NRC 2012a:** U.S. NRC, letter from Jason C. Paige to Mano Nazar, Florida Power and Light Company, "Turkey Point Units 3 and 4 - Issuance of Amendments Regarding Extended Power Uprate (TAC Nos. ME4907 and ME4908)," dated June 15, 2012. ADAMS Accession No. ML11293A359.

77. **NRC 2012b:** U.S. NRC, letter from Tracy J. Orf to Mano Nazar, Florida Power and Light Company, "St. Lucie Plant, Unit 2 - Issuance of Amendment Regarding Extended Power Uprate (TAC No. ME5843)," dated September 24, 2012. ADAMS Accession No. ML12268A132.

78. **NRC 2013:** U.S. NRC, Management Directive 8.4, "Management of Facility-Specific Backfitting and Information Collection," dated October 9, 2013. ADAMS Accession No. ML12059A460.

79. **NRC 2014a:** 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.

80. **NRC 2014b:** U.S. NRC, memorandum from Samuel Miranda to Christopher P. Jackson, "Making Non-Concurrence NCP-2013-04 Public," dated February 28, 2014. ADAMS Accession No. ML14063A174.

81. **NRC 2015a:** U.S. NRC, Draft Revision 1 to RIS 2005-29, "Anticipated Transients that Could Develop into More Serious Events," dated July 13, 2015. ADAMS Accession No. ML15014A469. (Also published at for public comment at 80 FR 42559.)

82. **NRC 2015b:** U.S. NRC, letter from Anne T. Boland to Bryan Hanson, Exelon Generation Company, LLC, "Braidwood Station, Units 1 and 2, and Byron Station, Unit Nos. 1 and 2 – Backfit Imposition Regarding Compliance with 10 CFR § 50.34(b), GDC 15, GDC 21, GDC 29, and Licensing Basis (TAC Nos. MF3206, MF3207, MF3208, and MF3209)," dated October 9, 2015. ADAMS Accession No. ML14225A871.

83. **NRC 2016a:** U.S. NRC, "Official Transcript of Proceedings – Public Meeting to Discuss Exelon Generating Company, LLC's Appeal of Compliance Backfit Affecting Byron and Braidwood Generating Stations," dated March 7, 2016. ADAMS Accession No. ML16070A364.

84. **NRC 2016b:** U.S. NRC, memorandum from Anthony T. Gody, Jr., to Marissa G. Bailey, "Input for Exelon Backfit Review Panel," dated March 21, 2016. ADAMS Accession No. ML16081A405 [non-public].

85. **NRC 2016c:** U.S. NRC, memorandum from Marissa G. Bailey to William M. Dean, "Backfit Review Panel Recommendation Regarding Exelon Appeal of Backfit Affecting Byron and Braidwood Stations Regarding Compliance with 10 CFR 50.34(b), GDC 15, GDC 21, GDC 29, and the Licensing Basis," dated March 25, 2016. ADAMS Accession No. ML16082A542 [non-public].

86. **NRC 2016d:** U.S. NRC, letter from William M. Dean to J. Bradley Fewell, Exelon Generation Company, LLC, "U.S. Nuclear Regulatory Commission Response to Backfit Appeal - Braidwood Station, Units 1 and 2, and Byron Station, Units 1 and 2," dated May 3, 2016. ADAMS Accession No. ML16095A204.

87. **NRC 2016e:** U.S. NRC, memorandum from Victor M. McCree to Gary M. Holahan, K. Steven West, Thomas G. Scarbrough, and Michael A. Spencer, "Charter for Backfit Appeal Review Panel Associated with Byron and Braidwood Compliance with 10 CFR 50.34(b), GDC 15, GDC 21, GDC 29, and the Licensing Basis," dated June 22, 2016. ADAMS Accession No. ML16173A311 [non-public].

88. **NRC 2016f:** U.S. NRC, "An Assessment of Core Damage Frequency for Byron/Braidwood Nuclear Power Plants Supporting Backfit Appeal Review Panel," dated August 11, 2016. ADAMS Accession No. ML16214A199 [non-public].

89. ~~**PG&E 1986:** Pacific Gas and Electric Company (PG&E), letter from H. Schierling to J.D. Shiffer, U.S. NRC, "[Diablo Canyon] Safety and Relief Valve Testing, NUREG-0737 Item II.D.1," dated January 27, 1986. [This document is not in ADAMS, but is available through the NRC Public Document Room using Accession No. 8602180450 and microfiche location 34630:271-299.]~~

89. **PG&E 1996:** ~~Pacific Gas and Electric Company~~PG&E, letter from Gregory M. Rueger to U.S. NRC, "Forwards completed Licensing Basis Impact Evaluation of FSAR Update change which contains SE performed IAW 10CFR50.59 re reanalysis of inadvertent ECCS actuation accident," dated August 13, 1996. [This document is not in ADAMS, but is available through the NRC Public Document Room using Accession No. 9608200112 and microfiche location 89419:294-322.]

90. **PG&E 2003:** Pacific Gas and Electric Company, letter from David H. Oatley to U.S. NRC, "Response to NRC Request for Additional Information Regarding License Amendment Request 01-018, 'Credit for Automatic Actuation of Pressurizer Power Operated Relief Valves; Pressurizer Safety Valve Loop Seal Temperature,'" dated November 21, 2003. ADAMS Accession No. ML033360735.

91. **TVA 1983:** Tennessee Valley Authority (TVA), letter from L.M. Mills to E. Adensam, U.S. NRC, enclosing "Watts Bar Nuclear Plant Units 1 and 2 Safety Valve Sizing," dated April 18, 1983. ADAMS Accession No. ML080360226.

92. **Union Electric 2000:** Union Electric Company, letter from Alan C. Passwater to U.S. NRC, "Revision to Technical Specifications 3.3.2, 3.4.10, and 3.4.11 – Pressurizer Safety Valves and PORVs," dated ~~September~~ May 25, 2000. ADAMS Accession No. ML003719636.

93. **VEPCO 2009:** Virginia Electric and Power Company (VEPCO), letter from L.N. Hartz to U.S. NRC, "North Anna Power Station Units 1 and 2 Updated Final Safety Analysis Report Revision 45," dated October 1, 2009. ADAMS Accession No. ML092810047.

94. **VEPCO 2015:** Virginia Electric and Power Company, letter from Gianna C. Clark to U.S. NRC, "North Anna Power Station Units 1 and 2 Updated Final Safety Analysis Report Revision 51," dated September 30, 2015. ADAMS Accession No. ML15296A098 [non-public].

95. **Westinghouse 1988:** Westinghouse, R.J. Dickinson and J.G. Bass, "Pressurizer Safety Relief Valve Operation for Water Discharge during a Feedwater Line Break," WCAP-11677, dated January 1988. (Submitted to the NRC as part of a May 8, 1989, response to a request for additional information related to Seabrook.) [This document is not in ADAMS, but is available through the NRC Public Document Room using Accession No. 8905120191 and microfiche location 49755:336 – 49756:017.]

96. **Westinghouse 1993:** Westinghouse, Nuclear Safety Advisory Letter (NSAL) 93-013, "Inadvertent ECCS Actuation at Power," dated June 30, 1993. [This document is not in public ADAMS, but is available through the NRC Public Document Room using Accession No. 9608200112 and microfiche location 89419:311-315, as well as in non-public ADAMS Accession No. ML052930330, pages 2-6 of file.]

97. **Westinghouse 1994:** Westinghouse, NSAL-93-013, Supplement 1, "Inadvertent ECCS Actuation at Power," dated October 28, 1994. ADAMS Accession No. ML050320117 [pages 9-15 of file].

98. **Westinghouse 2000:** Westinghouse, NSAL-00-013, "CVCS Modeling Assumption for Loss of Offsite Power Analyses," dated August 23, 2000. ADAMS Accession No. ML103200150 [pages 131-139 of file].

99. **Westinghouse 2007:** Westinghouse, NSAL-07-10, "Loss-of-Normal Feedwater Loss-of-Offsite AC Power Analysis PORV Modeling Assumptions," dated November 7, 2007. ADAMS Accession No. ML100140163 [pages 23-26 of file].

100. **Westinghouse 2011:** Westinghouse, "AP1000 Design Control Document," Revision 19, dated June 13, 2011. ADAMS Accession No. ML11171A500.

101. **WOG 1982:** Westinghouse Owners Group (WOG), letter from Oliver D. Kingsley, Alabama Power Company, to Harold R. Denton, U.S. NRC, "NUREG-0737, Item II.D.1, 'Pressurizer Safety Valve Operability,'" dated July 27, 1982. Forwards Westinghouse WCAP-10105, "Review of Pressurizer Safety Valve Performance as Observed in the EPRI Safety and Relief Valve Test Program," dated June 1982. [This document is not in ADAMS, but is available through the NRC Public Document Room using Accession No. 8208190307 and microfiche location 14387:189-301.]

MEMORANDUM TO: Victor M. McCree
Executive Director for Operations

FROM: Gary M. Holahan, Backfit Appeal Review Panel Chairman
Office of the Executive Director for Operations

K. Steven West, Deputy Director
Office of Nuclear Security and Incident Response

Thomas G. Scarbrough, Senior Mechanical Engineer
Office of New Reactors

Michael A. Spencer, Senior Attorney
Office of the General Counsel

Theresa Valentine Clark, Executive Technical Assistant
Office of the Executive Director for Operations

SUBJECT: BACKFIT APPEAL REVIEW PANEL FINDINGS ASSOCIATED WITH BYRON AND BRAIDWOOD COMPLIANCE WITH 10 CFR 50.34(b), GDC 15, GDC 21, GDC 29, AND THE LICENSING BASIS

In response to your memorandum of June 22, 2016, establishing a Backfit Appeal Review Panel (Agencywide Documents Access and Management System (ADAMS) Accession No. ML16173A311), the Panel undertook a review of the relevant documents in this case. This included the licensee and NRC staff letters related to the backfit; the 2001 power uprate and the 2004 valve setpoint license amendments (ADAMS Accession Nos. ML033040016 and ML042250531, respectively); and a June 16, 2016, letter from the Nuclear Energy Institute (NEI) supporting the Exelon backfit appeal (ADAMS Accession No. ML16208A008). The Panel also reviewed numerous other documents related to the topic of inadvertent operation of the emergency core cooling system (ECCS) and pressurizer safety valve performance.

In addition to the document review, the Panel had the benefit of meetings with the Office of Nuclear Reactor Regulation (NRR) (both the Division of Safety Systems and the Division of Engineering), the Office of the General Counsel, and the NRC Committee to Review Generic Requirements (CRGR). In addition, the Office of Nuclear Regulatory Research (RES) conducted an analysis that provided insights on the risk significance of the sequence at issue.

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301-415-1765

The Panel also shared its draft preliminary findings with NRR and OGC for comment. NRR provided comments, the consideration of which is reflected in the attached report. Both Exelon (Bradley Fewell, Senior Vice President of Regulatory Affairs) and NEI (Tony Pietrangelo, Senior Vice President and Chief Nuclear Officer) declined offers for a public meeting but indicated a willingness to provide information if the Panel identified the need. The Panel did not identify a need for additional information from either Exelon or NEI to complete its review, which is summarized below and documented in the attached report.

Based on its review, the Panel concludes that the NRC staff positions taken to support the compliance backfit finding represent new and different staff views on how to address potential pressurizer safety valve failures following water discharge. Although these staff positions are well-intentioned and conservative approaches that could provide additional safety margin, they do not provide a basis for a compliance backfit. In the absence of a failure of the pressurizer safety valve to reseat, the concerns articulated in the backfit related to event classification, event escalation, and compliance with 10 CFR 50.34(b) and General Design Criteria 15, 21, and 29 are no longer at issue.

The Panel notes, as did a member of the earlier NRR backfit appeal panel (ADAMS Accession No. ML16081A405), that the issue of pressurizer valve performance following water discharge appears to have generic applicability, and is not specific to Byron and Braidwood. The Panel believes that resolution of this issue would have benefited from consideration of the generic nature of the issue through the appropriate NRC processes.

Your June 22, 2016 memorandum asked the Panel to answer five questions. These questions and the Panel's responses follow:

1. **Were the approvals based on a mistake? If so, what was the mistake and what are the implications for Braidwood and Byron?**

   Answer: The 2001 and 2004 license amendments were based on reasonable and well-informed engineering judgment of the NRC staff, not a mistake.

2. **What is the known and established standard for water qualification of pressurizer safety valves?**

   Answer: The standard in place in 2001 and 2004 and at present is that failures of pressurizer safety valves to reclose need not be assumed to occur following water discharge if the likelihood is sufficiently small, based on well-informed staff engineering judgment.

3. **What is the known and established standard for progression of postulated events between categories of severity? Include a discussion of Regulatory Issue Summary 2005-29, "Anticipated Transients that Could Develop into More Serious Events," dated December 14, 2005 [ADAMS Accession No. ML051890212], and the draft Revision 1 that was issued for public comment in 2015 [ADAMS Accession No. ML15014A469].**

   Answer: For Byron and Braidwood, the standard for progression of postulated events between categories of severity is set forth in the Updated Final Safety Analysis Report (UFSAR), as described in the NRC staff's October 9, 2015, backfit imposition letter (ADAMS Accession No. ML14225A871). The Panel supports the NRC staff's view that non-escalation (from Condition II to Condition III or IV, as defined in American Nuclear Society Standard 51.1/N18.2-1973, "Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants," dated August 6, 1973) is a known and established standard applicable to Byron and Braidwood. However, this event progression standard does not establish specific standards for valve qualification. Therefore, it is not the basis for a compliance backfit given this set of facts. Regulatory Issue Summary 2005-29 and its draft Revision 1 do not alter this conclusion.

4. Does the current licensing basis for Braidwood and Byron comply with the applicable regulations? Is it adequate to provide protection to public health and safety?

Answer: The Panel concludes that the current licensing bases for Braidwood and Byron do comply with the applicable regulations based on the UFSAR analyses which the NRC staff found acceptable through a reasonable and technically sound evaluation using appropriate Commission safety standards. The Panel also concludes that there is reasonable assurance of adequate protection of the public health and safety.

5. Given that Exelon suggests that the NRC pursue a cost-justified substantial safety enhancement backfit, what is the contribution to overall plant risk of the current configuration at Braidwood and Byron?

Answer: The analysis performed for the Panel by RES provides insights on the risk significance of the sequence at issue. This analysis suggests that an inadvertent ECCS actuation sequence, assuming that pressurizer overfill leads to a small loss-of-coolant accident, contributes approximately 1 percent of the total internal events core damage frequency (CDF). If the backfit were implemented such that pressurizer overfill were always prevented, the CDF reduction is estimated at 1.5E-07 per year. If the PSVs are not assumed to always fail following water discharge (consistent with the NRC staff expert judgment in 2001) or a smaller improvement than a "perfect backfit" were considered, the risk-reduction benefit of implementing the backfit would be even smaller.~~Less conservative assumptions than these extremes would provide a smaller risk benefit through the backfit.~~

Comment [MAS]: I took this from the report. It's a clearer statement of how conservative the RES assumption was.

The Panel is aware of and sensitive to two important issues related to this question. First, NRR, not the appeal Panel, is responsible for any decisions on alternative application of the backfit rule to this issue. Second, the Panel does not wish to imply that "the contribution to plant risk" should be seen as the only measure of enhanced safety. For example, defense-in-depth has a recognized role and value in the regulatory process.

The Panel's findings therefore support the Exelon backfit appeal, and we recommend that you respond to NRR's repeal with a reversal of the October 9, 2015, backfit imposition. In addition, to address the generic nature of the issues described in the enclosed report, we recommend that you direct NRR to:

- verify (e.g., through letter, meeting, or owners group activity) that all pressurized-water reactors have resolved this technical issue in a reasonable manner, and

- re-evaluate the matters discussed in Regulatory Issue Summary 2005-29 and its draft Revision 1 through the appropriate generic process to avoid the inappropriate or inadvertent imposition of backfits.

Furthermore, in the course of its activities, the Panel has developed several insights relevant to the backfit process and the use of generic processes to address potential safety issues. The Panel plans to share these insights with the CRGR for its use in addressing your June 9, 2016, tasking related to implementation of agency backfitting and issue finality guidance. The Panel also identified other lessons from its review of the NRC evaluation of the performance of pressurizer safety valves for Braidwood, Byron, and other nuclear power plants that are identified in the attached report.

Finally, the Panel would like to recognize the valuable context and insights provided by NRR and OGC staff during this effort, and the timely and responsive efforts of RES in providing the comprehensive and useful risk analyses requested by the Panel.

The Panel is available to respond to any questions or provide any other assistance needed.

Enclosure: As stated

Panel plans to share these insights with the CRGR for its use in addressing your June 9, 2016, tasking related to implementation of agency backfitting and issue finality guidance. The Panel also identified other lessons from its review of the NRC evaluation of the performance of pressurizer safety valves for Braidwood, Byron, and other nuclear power plants that are identified in the attached report.

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Enclosure: As stated

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ADAMS Accession Nos.: Package ML16xxxxxxx
Memorandum ML16xxxxxxx; Enclosure ML16xxxxxxx ***via email**

OFFICE	OEDO	OEDO	NSIR	NRO	OGC
NAME	TClark	GHolahan	KWest	TScarbrough	MSpencer
DATE					

OFFICIAL RECORD COPY

From:	Clark, Theresa
To:	Scarbrough, Thomas; Spencer, Michael; West, Steven; Holahan, Gary
Subject:	REVIEW/FYI: updated report files
Date:	Monday, August 22, 2016 5:41:33 PM
Attachments:	cover memo (MASTER).docx Backfit Appeal Panel Report (MASTER).docx

Thanks, Tom!

The attached updated version includes Tom's edits, as well as the remaining ones from Steve that I didn't get to this morning, and is tracked from the version I sent this morning (not from the beginning) for those of you who were already looking at that one.

Some comments for discussion tomorrow. Also, I'll get cracking on that abbreviations list :).

See you all tomorrow!

-----Original Message-----
From: Scarbrough, Thomas
Sent: Monday, August 22, 2016 12:19 PM
To: Clark, Theresa <Theresa.Clark@nrc.gov>
Cc: Spencer, Michael <Michael.Spencer@nrc.gov>; West, Steven <Steven.West@nrc.gov>; Holahan, Gary <Gary.Holahan@nrc.gov>
Subject: RE: My comments on Friday's clean master

Theresa,

In my attached markup of the Friday version of the report, I propose a few minor changes with comments in the margin.

My only significant suggestion regarding the report findings is my proposed ending to the sentence at the top of page 13 regarding the assumption that PSV failure following water discharge is a passive failure. I think we should add a provision to the end of the sentence that the licensee needs to justify that assumption (such as by EPRI testing).

The changes in the version provided with your e-mail this morning look fine with a few edits as follows:

- Footnote 13 has an extra "not" in the sentence.
- On page 4 in the second line of the second paragraph, the word "depended" appears misspelled.
- On page 8 in the last paragraph, the word "panel" should be capitalized.
- On page 18 in the first full paragraph, the second sentence should use "had" rather than "would" before "conducted"
- On page 21 in the last full paragraph, it appears that the last sentence should specify "prevent" rather than "ensure" pressurizer overfilling.
- On page 23 in the first paragraph, the last sentence has an extra "not"

Thanks.
Tom

-----Original Message-----
From: Clark, Theresa
Sent: Monday, August 22, 2016 8:44 AM
To: West, Steven <Steven.West@nrc.gov>; Holahan, Gary <Gary.Holahan@nrc.gov>
Cc: Spencer, Michael <Michael.Spencer@nrc.gov>; Scarbrough, Thomas <Thomas.Scarbrough@nrc.gov>
Subject: RE: My comments on Friday's clean master

Thank you, Steve!

I put in your comments through Appendix A and will be out of the office at an OEDO meeting most of the rest of the day. I'll get the rest in as soon as I can and resend.

Note that I put a few comments in the margin, mostly for Steve's awareness, but one for Michael to check and one as a placeholder based on an email from Steve (will address more later).

-----Original Message-----
From: West, Steven
Sent: Sunday, August 21, 2016 7:35 PM
To: Clark, Theresa <Theresa.Clark@nrc.gov>; Holahan, Gary <Gary.Holahan@nrc.gov>
Cc: Spencer, Michael <Michael.Spencer@nrc.gov>; Scarbrough, Thomas <Thomas.Scarbrough@nrc.gov>
Subject: My comments on Friday's clean master

Any thoughts on meeting again before Wed?

MEMORANDUM TO: Victor M. McCree
Executive Director for Operations

FROM: Gary M. Holahan, Backfit Appeal Review Panel Chairman
Office of the Executive Director for Operations

K. Steven West, Deputy Director
Office of Nuclear Security and Incident Response

Thomas G. Scarbrough, Senior Mechanical Engineer
Office of New Reactors

Michael A. Spencer, Senior Attorney
Office of the General Counsel

Theresa Valentine Clark, Executive Technical Assistant
Office of the Executive Director for Operations

SUBJECT: BACKFIT APPEAL REVIEW PANEL FINDINGS ASSOCIATED WITH BYRON AND BRAIDWOOD COMPLIANCE WITH 10 CFR 50.34(b), GDC 15, GDC 21, GDC 29, AND THE LICENSING BASIS

In response to your memorandum of June 22, 2016, establishing a Backfit Appeal Review Panel (Agencywide Documents Access and Management System (ADAMS) Accession No. ML16173A311), the Panel undertook a review of the relevant documents in this case. This included the licensee and NRC staff letters related to the backfit; the 2001 power uprate and the 2004 valve setpoint license amendments (ADAMS Accession Nos. ML033040016 and ML042250531, respectively); and a June 16, 2016, letter from the Nuclear Energy Institute (NEI) supporting the Exelon backfit appeal (ADAMS Accession No. ML16208A008). The Panel also reviewed numerous other documents related to the topic of inadvertent operation of the emergency core cooling system (ECCS) and pressurizer safety valve performance.

In addition to the document review, the Panel had the benefit of meetings with the Office of Nuclear Reactor Regulation (NRR) (both the Division of Safety Systems and the Division of Engineering), the Office of the General Counsel, and the NRC Committee to Review Generic Requirements (CRGR). In addition, the Office of Nuclear Regulatory Research (RES) conducted an analysis that provided insights on the risk significance of the sequence at issue.

CONTACT: Gary M. Holahan, OEDO
301-415-1765

The Panel also shared its draft preliminary findings with NRR and OGC for comment. NRR provided comments, the consideration of which is reflected in the attached report. Both Exelon (Bradley Fewell, Senior Vice President of Regulatory Affairs) and NEI (Tony Pietrangelo, Senior Vice President and Chief Nuclear Officer) declined offers for a public meeting but indicated a willingness to provide information if the Panel identified the need. The Panel did not identify a need for additional information from either Exelon or NEI to complete its review, which is summarized below and documented in the attached report.

Based on its review, the Panel concludes that the NRC staff positions taken to support the compliance backfit finding represent new and different staff views on how to address potential pressurizer safety valve failures following water discharge. Although these staff positions are well-intentioned and conservative approaches that could provide additional safety margin, they do not provide a basis for a compliance backfit. In the absence of a failure of the pressurizer safety valve to reseat, the concerns articulated in the backfit related to event classification, event escalation, and compliance with 10 CFR 50.34(b) and General Design Criteria 15, 21, and 29 are no longer at issue.

The Panel notes, as did a member of the earlier NRR backfit appeal panel (ADAMS Accession No. ML16081A405), that the issue of pressurizer valve performance following water discharge appears to have generic applicability, and is not specific to Byron and Braidwood. The Panel believes that resolution of this issue would have benefited from consideration of the generic nature of the issue through the appropriate NRC processes.

Your June 22, 2016 memorandum asked the Panel to answer five questions. These questions and the Panel's responses follow:

1. **Were the approvals based on a mistake? If so, what was the mistake and what are the implications for Braidwood and Byron?**

   Answer: The 2001 and 2004 license amendments were based on reasonable and well-informed engineering judgment of the NRC staff, not a mistake.

2. **What is the known and established standard for water qualification of pressurizer safety valves?**

   Answer: The standard in place in 2001 and 2004 and at present is that failures of pressurizer safety valves to reclose need not be assumed to occur following water discharge if the likelihood is sufficiently small, based on well-informed staff engineering judgment.

3. **What is the known and established standard for progression of postulated events between categories of severity? Include a discussion of Regulatory Issue Summary 2005-29, "Anticipated Transients that Could Develop into More Serious Events," dated December 14, 2005 [ADAMS Accession No. ML051890212], and the draft Revision 1 that was issued for public comment in 2015 [ADAMS Accession No. ML15014A469].**

   Answer: For Byron and Braidwood, the standard for progression of postulated events between categories of severity is set forth in the Updated Final Safety Analysis Report (UFSAR), as described in the NRC staff's October 9, 2015, backfit imposition letter (ADAMS Accession No. ML14225A871). The Panel supports the NRC staff's view that non-escalation (from Condition II to Condition III or IV, as defined in American Nuclear Society Standard 51.1/N18.2-1973, "Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants," dated August 6, 1973) is a known and established standard applicable to Byron and Braidwood. However, this event progression standard does not establish specific standards for valve qualification. Therefore, it is not the basis for a compliance backfit given this set of facts. Regulatory Issue Summary 2005-29 and its draft Revision 1 do not alter this conclusion.

4. Does the current licensing basis for Braidwood and Byron comply with the applicable regulations? Is it adequate to provide protection to public health and safety?

Answer: The Panel concludes that the current licensing bases for Braidwood and Byron do comply with the applicable regulations based on the UFSAR analyses which the NRC staff found acceptable through a reasonable and technically sound evaluation using appropriate Commission safety standards. The Panel also concludes that there is reasonable assurance of adequate protection of the public health and safety.

5. Given that Exelon suggests that the NRC pursue a cost-justified substantial safety enhancement backfit, what is the contribution to overall plant risk of the current configuration at Braidwood and Byron?

Answer: The analysis performed for the Panel by RES provides insights on the risk significance of the sequence at issue. This analysis suggests that an inadvertent ECCS actuation sequence, assuming that pressurizer overfill leads to a small loss-of-coolant accident, contributes approximately 1 percent of the total internal events core damage frequency (CDF). If the backfit were implemented such that pressurizer overfill were always prevented, the CDF reduction is estimated at 1.5E-07 per year. If the PSVs are not assumed to always fail following water discharge (consistent with the NRC staff expert judgment in 2001) or a smaller improvement than a "perfect backfit" were considered, the risk-reduction benefit of implementing the backfit would be even smaller.~~Less conservative assumptions than these extremes would provide a smaller risk benefit through the backfit.~~

Comment [MAS]: I took this from the report. It's a clearer statement of how conservative the RES assumption was.

The Panel is aware of and sensitive to two important issues related to this question. First, NRR, not the appeal Panel, is responsible for any decisions on alternative application of the backfit rule to this issue. Second, the Panel does not wish to imply that "the contribution to plant risk" should be seen as the only measure of enhanced safety. For example, defense-in-depth has a recognized role and value in the regulatory process.

The Panel's findings therefore support the Exelon backfit appeal, and we recommend that you respond to NRR's repeal with a reversal of the October 9, 2015, backfit imposition. In addition, to address the generic nature of the issues described in the enclosed report, we recommend that you direct NRR to:

- verify (e.g., through letter, meeting, or owners group activity) that all pressurized-water reactors have resolved this technical issue in a reasonable manner, and

- re-evaluate the matters discussed in Regulatory Issue Summary 2005-29 and its draft Revision 1 through the appropriate generic process to avoid the inappropriate or inadvertent imposition of backfits.

Furthermore, in the course of its activities, the Panel has developed several insights relevant to the backfit process and the use of generic processes to address potential safety issues. The Panel plans to share these insights with the CRGR for its use in addressing your June 9, 2016, tasking related to implementation of agency backfitting and issue finality guidance. The Panel also identified other lessons from its review of the NRC evaluation of the performance of pressurizer safety valves for Braidwood, Byron, and other nuclear power plants that are identified in the attached report.

Finally, the Panel would like to recognize the valuable context and insights provided by NRR and OGC staff during this effort, and the timely and responsive efforts of RES in providing the comprehensive and useful risk analyses requested by the Panel.

The Panel is available to respond to any questions or provide any other assistance needed.

Enclosure: As stated

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Enclosure: As stated

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ADAMS Accession Nos.: Package ML16xxxxxxx
Memorandum ML16xxxxxxx; Enclosure ML16xxxxxxx ***via email**

OFFICE	OEDO	OEDO	NSIR	NRO	OGC
NAME	TClark	GHolahan	KWest	TScarbrough	MSpencer
DATE					

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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

Theresa V. Clark

August XX, 2016

ADAMS Accession No. MLXXXXXXXXX

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1 BACKGROUND

On June 22, 2016,[1] in accordance with NRC Management Directive (MD) 8.4,[2] the NRC Executive Director for Operations (EDO) established a Backfit Appeal Review Panel (Panel) to review the appeal by Exelon Generation Company, LLC (Exelon or the licensee) of the U.S. Nuclear Regulatory Commission (NRC) staff's determination that a backfit is necessary at Byron Station, Units 1 and 2 (Byron) and Braidwood Station, Units 1 and 2, as well as the NRC staff's application of the compliance backfit exception provided in Title 10 of the *Code of Federal Regulations* (10 CFR), Section 50.109, "Backfitting."

This backfit determination is documented in an October 9, 2015, letter (referred to as the Backfit Letter).[3] The letter describes the NRC staff's review of licensing basis documents for Byron and Braidwood. The NRC staff determined that Byron and Braidwood were not in compliance with the plant-specific design bases and several NRC regulations:

- General Design Criterion (GDC) 15, "Reactor coolant system design," in 10 CFR Part 50, Appendix A, "General Design Criteria for Nuclear Power Plants"
- GDC 21, "Protection system reliability and testability"
- GDC 29, "Protection against anticipated operational occurrences"
- Paragraph (b) of 10 CFR 50.34, "Contents of applications; technical information"

Specifically, the NRC staff determined that Byron and Braidwood do not comply with provisions in American Nuclear Society (ANS) Standard 51.1/N18.2-1973[4] for ensuring that ANS Condition II events[5] do not progress to more serious ANS Condition III events following water discharge[6] through certain valves. The NRC staff acknowledged that the NRC staff position differed from a previous staff position documented in a May 4, 2001, safety evaluation (SE) supporting a stretch power uprate (referred to as the Uprate SE).[7] However, the NRC staff determined that the backfitting was justified under the compliance exception in 10 CFR 50.109(a)(4)(i). The NRC staff directed the licensee to take action to resolve the non-compliance.

On December 8, 2015, the licensee appealed the NRC staff's decision to the Director of the Office of Nuclear Reactor Regulation (NRR), stating its disagreement with the NRC's conclusion that the compliance exception to the backfit rule applied in this case, while noting that the NRC staff had twice approved the underlying analysis.[8] The approvals referenced by the licensee

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[1] NRC 2016e (Author and year citations in footnotes refer to the designation of references in Appendix D to this report.)
[2] NRC 2013
[3] NRC 2015b – referred to as the Backfit Letter in the remainder of the report
[4] ANS 1973
[5] Specifically, inadvertent operation of the emergency core cooling system, malfunction of the chemical and volume control system, and inadvertent opening of a pressurizer safety or relief valve.
[6] For consistency in this report, the Panel uses the phrase "water discharge" rather than "water relief" or "liquid discharge" (except in direct quotes), as this is the phrase used in the Westinghouse documents that raised the issue addressed in this report.
[7] NRC 2001b – referred to as the Uprate SE in the remainder of the report
[8] Exelon 2015 – referred to as the NRR Appeal in the remainder of the report

were an August 26, 2004, license amendment associated with pressurizer safety valve (PSV) setpoints[9] and the above-referenced Uprate SE. In a letter dated May 3, 2016, the NRC responded to the licensee's appeal and reaffirmed its decision that the backfit per the compliance exception provisions of 10 CFR 50.109(a)(4)(i) is appropriate.[10]

On June 2, 2016, the licensee again appealed the NRC staff's decision, this time to the EDO.[11] The purpose of this report by the Backfit Appeal Review Panel is to provide information and recommendations to support the EDO's decision on the appeal.

1.1 Conduct of the Panel's Review

In order to establish a technically sound, well informed, and legally defensible basis for its recommendations, the Backfit Appeal Review Panel undertook a review of the relevant documents in this case. This included the licensee and NRC staff letters mentioned above; the Uprate SE and the Setpoint SE; and a June 16, 2016, letter from the Nuclear Energy Institute (NEI)[12] supporting the EDO Appeal. The Panel also reviewed many other related documents, which fall into five broad categories:

- The Backfit Rule (10 CFR 50.109), related court actions, and Commission and staff guidance on application of the Backfit Rule

- Docketed communications for Byron and Braidwood from 1982 to the present, including license amendment requests (LARs) by the licensee, NRC-issued license amendments, NRC requests for additional information (RAIs), licensee responses, meeting summaries, NRC SEs, and the licensee's Updated Final Safety Analysis Report (UFSAR)[13]

- NRC guidance relevant to the analysis of inadvertent operation of the emergency core cooling system (IOECCS) events over the period of 1981 to the present, including Standard Review Plan (SRP) Section 15.0, Sections 15.5.1 – 15.5.2, and Section 15.6.1[14]

- Westinghouse Nuclear Safety Advisory Letter (NSAL) 93-013[15] and its Supplement 1[16], as well as docketed communications regarding actions taken by other licensees in response to Westinghouse NSAL-93-013

- The history of NRC and industry activities related to power operated relief valves (PORVs), their block valves, and PSVs (including Three Mile Island (TMI) Action Plan Items II.D.1, II.D.3, II.G.1, and II.K.3 as documented in NUREG-0737[17], as well as

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[9] NRC 2004b – referred to as the Setpoint SE in the remainder of the report
[10] NRC 2016d – referred to as NRR Appeal Decision in the remainder of the report
[11] Exelon 2016a – referred to as EDO Appeal in the remainder of the report
[12] NEI 2016
[13] Exelon 2002 and Exelon 2014 (The Panel reviewed other revisions as well, but ~~not~~ they are not included in Appendix D as they are not referenced in this report.)
[14] NRC 1981a, NRC 1981b, NRC 1981c, NRC 2007a, NRC 2007b, and NRC 2007c
[15] Westinghouse 1993
[16] Westinghouse 1994
[17] NRC 1980c – referred to as the TMI Action Plan in the remainder of the report; lessons learned from TMI were also presented in NUREG-0578 (NRC 1979a), NUREG-0585 (NRC 1979b), and NUREG-0660 (NRC 1980a)

Generic Letter 89-10[18] and its supplements), Electric Power Research Institute (EPRI) valve testing, and operating experience (NUREG/CR-7037[19])

In addition to the document review, the Panel had the benefit of meetings with NRR (both the Division of Safety Systems and the Division of Engineering), the Office of the General Counsel, and the NRC Committee to Review Generic Requirements (CRGR). Both Exelon (Bradley Fewell, Senior Vice President of Regulatory Affairs) and NEI (Tony Pietrangelo, Senior Vice President and Chief Nuclear Officer) declined offers for a public meeting, but indicated a willingness to provide information if the Panel identified the need. The Panel did not identify a need for additional information from either Exelon or NEI to complete the review documented in this report.

At the request of the Panel, the Office of Nuclear Regulatory Research (RES) conducted risk analyses using the NRC's Standardized Plant Analysis Risk model for Byron Unit 1.[20] These analyses informed the Panel's response to the question from the EDO regarding the risk significance of the relevant accident sequences.

Given that the Backfit Rule creates a structured process for changes to previous NRC staff positions—in effect, placing the burden of proof on the NRC staff—the Panel determined that this level of historical review and staff interaction was necessary to provide context for consideration of the validity of the backfit.

Comment [CT]: I'm open to comment or deletion on this, but this is my attempt to address Steve's comment about the first (NRR) panel review scope.

1.2 Proposed Compliance Backfit and Exelon Appeals

In the Backfit Letter, the NRC staff informed Exelon that it had determined that Byron and Braidwood are not in compliance with GDCs 15, 21, and 29; 10 CFR 50.34(b); and the plant-specific design bases that were expected to demonstrate there will be no progression of ANS Condition II events to ANS Condition III events. The NRC staff stated that based on its review of Byron and Braidwood UFSAR Sections 15.5.1, 15.5.2, and 15.6.1, the UFSAR predicts water discharge through a valve that is not "qualified" for water discharge. Therefore, the NRC staff concluded that the UFSAR does not contain analyses that demonstrate that the plants' structures, systems, and components (SSCs) meet the design criteria for ANS Condition II events as stated in Byron and Braidwood UFSAR Section 15.0.1.2. Based on the SE attached to its letter,[21] the NRC staff found that the licensee must take action to resolve the non-compliance.

The Backfit SE addressed three accident analyses in Chapter 15 of the Byron and Braidwood UFSAR: (1) IOECCS; (2) chemical and volume control system (CVCS) malfunction that increases reactor coolant inventory; and (3) inadvertent opening of a pressurizer safety or relief valve (IOPORV). The NRC staff noted that each ANS Condition II event must be shown to meet the following:

1. no fuel damage,
2. no overpressure of the reactor coolant system (RCS) or main steam system, and

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[18] NRC 1989
[19] NRC 2011
[20] NRC 2016f
[21] Referred to as the Backfit SE in the remainder of the report.

3. no progression into an event of a more serious category without another independent fault.

Comment [CT]: Comment placeholder – think about more discussion on scope of NRR appeal.

Regarding an IOECCS, the NRC staff stated in Section 3.1.2.1 of the Backfit SE that use of the block valve to isolate a stuck-open PORV was unacceptable. The NRC staff stated that Westinghouse recommended this approach in 1993, and that the NRC staff rejected this approach in 2005 (RIS 2005-29[22]).

In Section 3.1.2.4 of the Backfit SE, the NRC staff stated that the Byron and Braidwood IOECCS analysis dependded on water discharge through the PSVs. The NRC staff faulted the licensee for "not appl[ying] the single-failure assumption" and stated that the following information was necessary to support water qualification of the PSVs:

1. In accordance with the American Society of Mechanical Engineers (ASME) *Boiler and Pressure Vessel Code* (BPV Code), Section III, provide the original Overpressure Protection Report defining operating conditions and required relief capacities, and manufacturer's certification and test results

2. In accordance with the ASME *Code for Operation and Maintenance of Nuclear Power Plants* (OM Code), provide inservice test history for PSVs, including water and steam tests, or provide correlation test for alternative test fluid.

Regarding a CVCS malfunction, the NRC staff stated in Section 3.2 of the Backfit SE that the licensee had not provided an analysis for the CVCS malfunction that increases reactor coolant inventory that demonstrated the plants' ability to meet the requirements of an ANS Condition II event.

Regarding an IOPORV, the NRC staff stated in Section 3.3 of the Backfit SE that the licensee had not provided an analysis for the IOPORV that extends long enough into the transient to demonstrate the event would not transition from an ANS Condition II event to an ANS Condition III event.

In the Backfit SE, the NRC staff referenced Millstone[23] and Callaway[24] license amendments as examples of licensees upgrading PORVs for water discharge; a Beaver Valley extended power uprate (EPU) license amendment[25] as an example of qualifying PORVs for water discharge; and Turkey Point[26] and St. Lucie Unit 2[27] EPU amendments as additional precedent in support of the backfit decision.

In the NRR Appeal, Exelon asserted that the NRC had not justified invoking the compliance exception to the backfit rule. Exelon stated that the NRC approved its IOECCS analysis in both the Uprate SE and the Setpoint SE.

In the NRR Appeal Decision, the NRC staff stated that the previous NRC approvals in 2001 and 2004 were inconsistent with the Agency's general position on the known and established standard at issue—in this case, the progression of ANS Condition II events to higher level

[22] NRC 2005b
[23] NRC 1998
[24] NRC 2000
[25] NRC 2006
[26] NRC 2012a
[27] NRC 2012b

events. The NRC staff stated that the fact that the NRC staff were aware of references to EPRI reports on the ability of these non-water qualified PSVs to reseat in certain circumstances was not sufficient to support the licensee's position on the compliance backfit.

In the EDO Appeal, Exelon stated that the NRC had misidentified the "known and established standard" at issue as the prohibition of ANS Condition II events progressing to ANS Condition III events. Exelon asserted that the standard in question concerns what is necessary to "qualify" valves for water discharge. Exelon contended that this standard was the EPRI testing and analysis, and that the NRC agreed that Byron and Braidwood met this standard. Exelon also contended that the change in NRC staff position on prior approvals was not a mistake of fact, but rather a new or modified interpretation of compliance with NRC requirements, for which use of the compliance exception provided for in the Backfit Rule was not appropriate.

1.3 Backfit Rule and the Compliance Exception

Backfitting is defined by 10 CFR 50.109(a) as:

> … the modification of or addition to systems, structures, components, or design of a facility; or the design approval or manufacturing license for a facility; or the procedures or organization required to design, construct or operate a facility; any of which may result from a new or amended provision in the Commission's regulations or the imposition of a regulatory staff position interpreting the Commission's regulations that is either new or different from a previously applicable staff position … .

Unless one of three specified exceptions apply, the NRC may impose a backfit only if it performs a backfit analysis in accordance with 10 CFR 50.109(a)(2) and determines in accordance with 10 CFR 50.109(a)(3) "that there is a substantial increase in the overall protection of the public health and safety or the common defense and security to be derived from the backfit and that the direct and indirect costs of implementation for that facility are justified in view of this increased protection."

Section 50.109(a)(4) sets forth the three exceptions to the requirements of 10 CFR 50.109(a)(2) and (a)(3). The first exception, the compliance exception, applies if the "modification is necessary to bring a facility into compliance with a license or the rules or orders of the Commission, or into conformance with written commitments by the licensee." The second and third exceptions relate to actions necessary to ensure adequate protection or to actions that involve defining or redefining adequate protection.

The Commission explained its intended application of the compliance exception in the Statements of Consideration (SOC) accompanying the 1985 final rule amending 10 CFR 50.109:[28]

> The compliance exception is intended to address situations in which the licensee has failed to meet known and established standards of the Commission because of omission or mistake of fact. It should be noted that new or modified interpretations of what constitutes compliance would not fall within the exception and would require a backfit analysis and application of the standard.

[28] NRC 1985, at 38103

In the same SOC, the Commission acknowledged that staff interpretations of rules are not legally binding, but the Commission also stated that "staff interpretations of broadly stated rules are often necessary to give a rule effect and in some instances may be a causal factor in initiating a backfit."[29]

By its terms, the compliance exception applies to actions necessary for compliance with rules, licenses, and orders, or for conformance with written commitments.[30] Also, the Commission explicitly acknowledged the importance of staff interpretations of rules in the regulatory process. Thus, the Panel understands the term "known and established standard" to include standards established in rules, licenses, orders, and written commitments, and NRC interpretations of rules. Some standards may be broad-based, while others may apply only to a limited number of plants. As stated in NUREG-1409, "[i]nformal or formal communications to one licensee are not official positions to all licensees. ... Orders, licenses, and written commitments are applicable only to a particular licensee."

The failure to meet a known and established standard is grounds for a compliance backfit if this failure is due to "omission or mistake of fact." Thus, if a licensee obtains NRC approval of an alternative to a specific standard set forth in guidance, that standard and guidance could not be used to support a compliance backfit unless the NRC's approval of the alternative was based on an omission or mistake of fact. "Known and established standards" are to be distinguished from "new or modified interpretations of what constitutes compliance," which do not fall within the compliance exception. The Panel understands the term "new or modified interpretations" to include situations where the NRC staff has, in effect, "changed its mind" on how to interpret the language of a requirement or on how much assurance is necessary to conclude that the requirement is met. Levels of assurance might be established in terms such as acceptable probabilities or consequences, conservative assumptions, or sufficient margin.

Additional background information on the Backfit Rule and the compliance exception is provided in Appendix A to this report.

1.4 A Brief History of Pressurizer Valve Issues

Appendix B to this report provides a summary of the NRC and industry's testing, evaluation, and other consideration of PORVs and PSVs since the TMI Unit 2 (TMI-2) accident in 1979. This historical review provides context for discussion of valve "qualification" in the Backfit SE. It also provides the basis for the Panel's conclusions regarding the "known and established standard" for "qualification" in the context of TMI Action Plan Item II.D.1 and subsequent activities, as well as how it should be interpreted in the Byron and Braidwood licensing basis.

In light of the NRC staff's assertion that the licensee had not applied the "single-failure assumption" as noted above, the Panel also considered the applicability of the single failure

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[29] NRC 1985, at 38102. The 1985 backfit rule was vacated by a Federal court on grounds unrelated to the compliance backfit exception. See *Union of Concerned Scientists v. U.S. Nuclear Regulatory Com'n*, 824 F.2d 108, 119-20 (1987). In 1988, the Commission amended the backfit rule (NRC 1988b) to address the court's concerns, but did not change the 1985 rule's compliance exception provision. Thus, the quoted statements from the 1985 rule are the applicable expression of Commission intent regarding compliance backfits.

[30] NUREG-1409 (NRC 1990c) defines written commitments broadly to include the "final safety analysis report, licensee event reports, and docketed correspondence, including responses to NRC bulletins, generic letters, inspection reports, or notices of violation and confirmatory action letters."

criterion to PSVs. The Panel expended considerable effort in searching for an answer to what appears to be a simple question: "Are PSVs active components subject to the single failure criterion, or are they passive components exempt from the single failure criterion?" NRR staff have taken the position that PSVs have consistently been treated as active components.

Comment [CT]: Differing views on this topic... but as I am of the "a historical" persuasion personally, I'm taking Tom's edit ☺.

In the Panel's evaluation of the treatment of PSV failure potential (Section 3 below), a historical perspective is provided. In general, the Panel found that the classification of a component as "active" or "passive" depends on its design, application, and function. For example, passive components almost always do not need external power; usually do not need an external actuator (e.g., signal)[31]; sometimes do not involve any mechanical motion (e.g., movement of a valve disc)[32]; and sometimes do not involve any motion, either fluid or mechanical (e.g., piping). While it does not represent formal NRC guidance, additional views on passive components are included in International Atomic Energy Agency (IAEA) TECDOC-1624.[33] This document states that "[s]afety related terms such as passive and inherent safety have been widely used, particularly with respect to advanced nuclear plants, generally without definition and sometimes with definitions inconsistent with each other." This guidance further defines four levels of "passivity" to "help eliminate confusion and misuse of the terms by members of the nuclear community." In addition, SECY-05-0138[34] also acknowledged and discussed inconsistencies in the use and application of the term "passive." Additional consideration of this topic by the Panel is documented in Section 3.10 below.

The introduction to the GDCs and the related footnote define the applicability of the single failure criterion in terms of electrical versus fluid systems, and active versus passive components. Neither the GDCs nor NRC guidance define which characteristics of passive components are necessary to make a component exempt from the single failure criterion. Some examples are clear: pipes are passive components and pumps and motor-operated valves that operate to perform their safety functions are active components. As discussed in Section 3.6 below, check valves might be classified as active or passive components depending on certain specific considerations.

With respect to PSVs, the ASME BPV Code applicable to Byron and Braidwood includes requirements for overpressure protection that relate to the single failure criterion through several specific design and construction requirements. As a result, the PSVs are conservatively sized with sufficient margin to accommodate a single failure although the single failure criterion is almost never explicitly discussed or applied in accident analyses. The Byron and Braidwood UFSAR states that "adequate overpressurization protection is provided by the three installed safety valves." Neither the UFSAR system descriptions nor the safety analyses provide detailed discussions of potential PSV failures or their consequences. The principal discussion of

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[31] For example, SECY-77-439 (NRC 1977) states: "Examples [of passive failures in fluid systems] include the failure of a simple check valve to move to its correct position when required, the leakage of fluid from failed components, such as pipes and valves—particularly through a failed seal at a valve or pump—or line blockage. Motor-operated valves which have the source of power locked out are allowed to be treated as passive components."

[32] For example, NUREG-1800 (NRC 2001c) states that "'[p]assive' structures and components, for the purpose of the license renewal rule, are those that perform an intended function ... without moving parts or without a change in configuration or properties ... 'passive' may also be interpreted to include structures and components that do not display 'a change of state.'"

[33] IAEA 2009

[34] NRC 2005a

potential PSV failures in the accident analyses occurs in the evaluation of an inadvertent opening of a PSV in UFSAR Section 15.6.1.

Most relevant for the current issue, the Byron and Braidwood UFSAR analyses of overpressure events (e.g., loss of load, loss of feedwater) do not apply the single failure criterion to cause a PSV to stick open (i.e., fail to reseat) when opening on steam flow. In addition, the UFSAR Feedwater System Pipe Break analysis (Chapter 15.2.8) does not apply the single failure criterion to cause a PSV to stick open either during steam discharge or during water discharge. A survey of other Westinghouse-designed plants showed that this treatment of PSV valve performance during anticipated operational occurrences (AOOs, similar to ANS Condition II events) and postulated accidents (similar to ANS Condition IV events) has been consistent and without any identified exceptions.[35]

1.5 History and Review of Westinghouse NSAL and Related Activities

Appendix C to this report provides the Panel's review of the issues identified by Westinghouse in NSAL-93-13 and its Supplement 1, how various licensees responded to these issues, and how the NRC was involved in reviewing and approving these actions. This review provides the basis for the Panel's conclusions related to the approach taken by Byron and Braidwood to address these issues in their licensing basis, as well as on the "known and established standard" for event escalation from ANS Condition II to ANS Condition III, referred to hereafter as the "non-escalation position."

2 SUMMARY OF THE APPEAL REVIEW PANEL FINDINGS

For the reasons provided in Section 3, the Panel concluded that in 2001 and 2004 and at present, the known and established standard of the Commission is 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. The Panel also concluded that, in preparing the Uprate SE and the Setpoint SE, the NRC staff exercised reasonable and well-informed engineering judgment when the NRC staff concluded that the PSVs were unlikely to stick open. The non-escalation position does not establish specific standards for valve qualification, so the non-escalation position, standing alone, provides no basis for rejecting the licensee's reliance on EPRI valve testing. Moreover, the Panel found that no mistake or error occurred in the licensee's or previous staff's reliance on the EPRI testing program that included an evaluation of water discharge through pressurizer valves.[36] Therefore, the Panel also concluded that the NRC staff's position on valve qualification in the Backfit SE is a new or modified interpretation of what constitutes compliance.

The P~~p~~anel also concluded that the issue of pressurizer valve performance following water discharge appears to have generic applicability, and is not specific to only Byron and Braidwood. The Panel believes that resolution of this issue would have benefited from consideration of the generic nature of the issue through the appropriate NRC processes. The Panel included additional information about this finding in Section 6 and Appendices B and C below.

[35] Examples include Watts Bar (NRC 1982 and TVA 1983), North Anna (NRC 1976), and AP1000 (Westinghouse 2011).
[36] "Pressurizer valves" is used in this report to refer to either PORVs or PSVs when discussing issues common to both types of valves.

3 DISCUSSION

The compliance exception to the Backfit Rule is intended to address failures to meet known and established Commission standards because of omission or mistake of fact. New or modified interpretations of what constitutes compliance do not fall within the exception. The Panel reviewed and evaluated the information referenced in this report to determine if, in 2001 and 2004, there was a known and established standard of the Commission relating to the potential for PSVs to fail following water discharge during IOECCS events.

In addition, the Panel considered the issue of "known and established standards of the Commission" as it relates to "event escalation." The NRR Appeal Decision stated that the Backfit SE "showed that the approvals at issue for Braidwood and Byron were inconsistent with the Agency's general position on the known and established standard at issue, in this case the progression of [ANS] Condition II events." The Panel recognizes that the non-escalation position, although not included in NRC regulations, is widely referenced in reactor licensing bases as an approach for addressing AOOs and postulated accidents as articulated in the GDCs. The non-escalation position is incorporated in Section 15.0.1.2 of the Byron and Braidwood UFSAR as "By definition, these faults (or events) do not propagate to cause a more serious fault, i.e., [ANS] Condition III or IV events."

Exelon and the Panel agree that the non-escalation position is now, and was in 2001 and 2004, a part of the licensing basis of both Byron and Braidwood. In addition, the Panel supports the NRC staff's view that non-escalation (from ANS Condition II to ANS Condition III or IV) is a known and established standard applicable to Byron and Braidwood. However, the Panel also agrees with Exelon that the fundamental issue is not the non-escalation position, as the NRC staff contends, but rather the appropriate standard for PSV water discharge. In the absence of a PSV failure to reseat, the concerns articulated by the NRC staff in the backfit related to event classification, event escalation, and compliance with 10 CFR 50.34(b) and GDCs 15, 21, and 29 would no longer be at issue.

The Panel's evaluation of the treatment of PSV failure potential includes an assessment of multiple relevant references, which are discussed chronologically in the sections that follow.

3.1 General Design Criteria (1971)

In 1971, the Atomic Energy Commission published the GDCs, which had been under development since 1965.[37] The introduction to 10 CFR Part 50, Appendix A addresses "Single Failure" in the section on Definitions and Explanations. The paragraph on single failures includes a footnote stating: "The conditions under which a single failure of a passive component in a fluid system should be considered in designing the system against a single failure are under development" (emphasis added).

3.2 Commission Paper on Single Failure (1977)

In response to several staff concerns and differing views on the subject of application of the single failure criterion, the Acting Director of NRR issued SECY-77-439 "[t]o inform the Commission of the present status and future use of the Single Failure Criterion as a tool in the reactor safety process."[38] In part, that paper addressed the application of the single failure

[37] AEC 1971
[38] NRC 1977

criterion to passive components in fluid systems, stating that "[a]pplication of the [single failure] concept is complicated by the interrelationships between the various fluid and electrical systems and their supporting auxiliaries in a nuclear power plant. Furthermore, there is a need to stipulate the events and associated assumptions which must be considered during application of the Single Failure Criterion."

SECY-77-439 specifically spoke to how "additional passive failures"—that is, failures in addition to the initiating event—had been and should be addressed, stating (with emphases added):

> During subsequent years [since the single failure footnote quoted above was published] staff assumptions regarding the nature of passive failures which should be considered have not been completely consistent and there has been some disagreement. However, on the basis of the licensing review experience accumulated in the period since 1969, it has been judged in most instances that the probability of most types of passive failures in fluid systems is sufficiently small that they need not be assumed in addition to the initiating failure in application of the Single Failure Criterion to assure safety of a nuclear power plant.

Furthermore, SECY-77-439 provides definitions and examples for distinguishing between active and passive failures. Among these examples, SECY-77-439 cites "the failure of a simple check valve to move to its correct position when required" as a passive failure. Of the examples cited in SECY-77-439, the check valve example is most similar from a mechanical perspective to the PSV failure addressed in the Backfit SE, as explained below in the discussion of SECY-94-084.

SECY-77-439 also stresses the use of engineering judgment relating to the probability of component failure and does not suggest that valve "certification" or "qualification" in accordance with ASME standards should be invoked as the basis for such decisions.

3.3 TMI Action Plan Item II.D.1 (1980)

As an element of the TMI Action Plan, the NRC staff required licensees to address the capability of relief and safety valves to perform their intended functions without failure. Specifically, Item II.D.1 states that "[p]ressurized-water reactor [PWR] and boiling-water reactor [BWR] licensees and applicants shall conduct testing to qualify the [RCS] relief and safety valves under expected operating conditions for design-basis transients and accidents." With reference to planned EPRI testing and other generic industry test programs, NUREG-0737 specified provisions for then-operating nuclear power plants and applicants for operating licenses and holders of construction permits to address the TMI Action Plan items, including Item II.D.1. NUREG-0737 stated, for the performance testing of relief and safety valves for Item II.D.1, that "[t]he testing should demonstrate that the valves will open and reclose under the expected flow conditions."

Although limited in scope, the EPRI test results did not identify any generic issues with PSVs or PORVs sticking open following water discharge. The NRC staff approvals summarized below show that the word "qualify" in this TMI Action Plan item was not intended to refer to ASME valve certification or qualification. Instead, "qualify" was used in a less formal sense to refer to a reasonable judgment that the valve would open to relieve pressure and then reliably reseat. As referenced in NUREG-0737, the EPRI test program was the widely used approach to address TMI Action Plan Item II.D.1 at PWR nuclear power plants. The Westinghouse Owners Group submitted WCAP-10105 to the NRC in 1982 to demonstrate the acceptability of the EPRI testing program for PSVs and PORVs in Westinghouse-designed PWRs.[39]

3.4 NRC Closure of TMI Action Plan Item II.D.1 for Byron and Braidwood (1988-1990)

A 1988 letter from the NRC staff to the licensee for Byron found the licensee's reliance on EPRI testing of PSVs to be acceptable.[40] The 1988 SE states that the test program was designed "[t]o reconfirm the integrity of the overpressure protection system and thereby assure that the [GDCs] are met." As discussed in Appendix B to this report, the 1988 SE described the NRC staff's evaluation of the PSVs and PORVs for feedwater line break accidents that would include water discharge, and determined that the EPRI tests were applicable to the Byron and Braidwood PSVs and PORVs. Based on the NRC staff and contractor review, the 1988 SE found that the performance of the PSVs and PORVs was acceptable based on the EPRI tests.

For the specific extended high pressure injection event, the 1988 SE states that water discharge through the PSVs and PORVs could be disregarded because of the long time available for operator action. However, the SE addressed water discharge through the PSVs and PORVs as part of the feedwater line break evaluation.

In the cover letter for the 1988 SE, the NRC staff states that the licensee should develop and adopt plant procedures to inspect the pressurizer valves after each lift involving loop seal or water discharge. The 1988 SE contains no reference to or suggestion of a need for certification of these valves in accordance with the ASME BPV Code for water discharge capability. In 1990, the NRC staff also found the use of the EPRI test program similarly acceptable for Braidwood.[41]

3.5 Westinghouse NSAL-93-013 and Supplement 1 (1993-1994)

In 1993, Westinghouse sent NSAL-93-013 to operating nuclear power plants in response to its discovery that potentially non-conservative assumptions had been used in the licensing analysis of the IOECCS event. Westinghouse recommended that licensees determine if their pressurizer safety relief valves (PSRVs)[42] "are capable of closing following discharge of subcooled water." Westinghouse noted that the PSRVs might have been designed or "qualified" to relieve subcooled water. Westinghouse also noted that "licensees may have qualified these valves in compliance to NUREG-0737, Item II.D.1." If the PSRVs were not designed or qualified for subcooled water discharge, Westinghouse recommended that licensees reevaluate the IOECCS event with three possible options of (1) reducing emergency core cooling system (ECCS) flow used in the safety analysis, (2) using a less restrictive operator response time, or (3) crediting the use of one or more PORVs to help mitigate the accident.

Later, in Supplement 1 to NSAL-93-013, Westinghouse alerted licensees to potential reduced time for operator action if a positive displacement pump (a typical component of the CVCS) were in service, and to the need to qualify the PSRVs and the piping downstream of the PSRVs and PORVs if water discharge from the pressurizer is predicted.

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[39] WOG 1982
[40] NRC 1988c, referred to as the 1988 SE
[41] NRC 1990a
[42] Westinghouse used the term PSRVs. The specific valves for Byron and Braidwood should be designated as "safety valves" or "pressurizer safety valves" as they are by the manufacturer, in the ASME BPV Code, and by the licensee. This difference in terminology is not significant to any of the findings or conclusions in this report.

Some licensees submitted license amendments that involved improvements to the PORVs and their circuitry to avoid water discharge through the PSVs (e.g., Salem[43], Millstone[44], Callaway[45], and Diablo Canyon[46]). The NRC staff review and approval of those proposed improvements relied on engineering judgment relative to the various test information and PORV circuitry upgrades described by individual licensees. The licensee for Byron and Braidwood submitted an LAR for similar PORV improvements,[47] but that request was later withdrawn.[48]

As indicated below, the Panel's sampling review found at least two plants, in addition to Byron and Braidwood, that chose to address this issue by crediting the capability of PSVs to relieve water, based on the EPRI testing performed in response to TMI Action Plan Item II.D.1.

3.6 Commission Paper on Passive Plant Designs (1994)

In 1994, in preparation for the design certification reviews of passive reactor designs (e.g., the Westinghouse Advanced Passive 1000 (AP1000) and the General Electric Economic Simplified Boiling-Water Reactor (ESBWR)), the NRC staff presented nine issues to the Commission for policy decisions.[49] Although PSV categorization and performance requirements were not explicitly addressed, the paper does include an issue on "Definition of Passive Failure" and an extensive discussion on whether check valves are passive or active components and how they should be addressed in current plants and future passive designs.

SECY-94-084 recognized the GDCs and SECY-77-439 as establishing long-standing requirements and guidance in this area. The paper acknowledged that the industry (including EPRI documents and ANSI/ANS 58.9[50]) have been inconsistent with respect to check valve failures, sometimes considering them as "active failures" and sometimes as "passive failures." In SECY-77-439, however, the NRC staff stated that the failure of a simple check valve to move to its correct position when required was a "passive failure." In addition, SECY-94-084 states that "[i]n licensing reviews, however, only on a long-term basis [e.g., long-term recirculation cooling following a loss of coolant accident (LOCA)] does the NRC staff consider passive failures in fluid systems as potential accident initiators in addition to initiating events." The paper also states that "[f]or current plants, the NRC staff normally treats check valves, except for those in containment isolation systems, as passive devices during transients or design-basis accidents."

Furthermore, SECY-94-084 states that "[r]edefining check valves as active components, subject to consideration for single active failures would cause these valves to be evaluated in a more stringent manner than that used in previous licensing reviews" (emphasis added). The NRC staff then recommended (and the Commission agreed[51]) that the NRC staff should "maintain the current licensing practice for passive component failures on the passive [advanced light water reactor] ALWR designs, and to redefine check valves, except for those whose proper function can be demonstrated and documented, in the passive safety systems as active components

[43] NRC 1997
[44] NRC 1998
[45] NRC 2000
[46] NRC 2004a
[47] ComEd 1998
[48] ComEd 1999
[49] NRC 1994a
[50] ANS 1981
[51] NRC 1994b

subject to single failure consideration." Therefore, the NRC's position on check valves was changed only for passive ALWR designs going forward.

Comment [CT]: Edited a bit from Tom's suggestion.

The Panel considered the opening function of check valves and PSVs to be similar in that they both open through the motion of the valve disk under differential pressure with no external signal or motive power. The Panel also recognized that the ambiguity with respect to "passive" versus "active" component definitions and nomenclature exists for safety valves. In addition, the passive or active classification of check valves or safety valves may differ based on design considerations, inservice testing, or accident analyses. For example, the PSVs and PORVs, as well as numerous check valves, are classified as active components in the Byron and Braidwood inservice testing programs. However, for purposes of applying the single failure criterion in the GDC context, the Panel concluded that it is appropriate to consider the potential failure of a PSV following water discharge as a passive failure~~,~~ (consistent with the treatment of check valve failures for the operating fleet), provided the licensee or applicant qualifies the performance of the PSV in an acceptable manner. In the case of Byron and Braidwood, the NRC staff accepted the EPRI testing associated with TMI Action Plan Item II.D.1 to provide this qualification.

3.7 Draft Standard Review Plan Revision (1996)

The 1996 draft revision to SRP Sections 15.5.1 – 15.5.2 on IOECCS and CVCS malfunctions includes extensive updates to the 1981 revision, but neither version includes any discussion, criteria, or guidance on applying ASME Code requirements to PSVs or on applying the single failure criterion or any other failure assumption to PSVs.[52]

3.8 Power Uprate Reviews and License Amendments (2001-2006)

As part of the 2001 power uprate review for Byron and Braidwood, the NRC staff approved the analysis of an IOECCS (UFSAR Section 15.5.1) that included pressurizer filling, PSV water discharge, ECCS termination, and PSV closure. In the Backfit SE, the NRC staff indicated that the 2001 license amendment was predicated on the NRC's mistaken (unsubstantiated) belief that the valves were ASME-qualified (certified). However, the Panel's review of the SE and associated RAIs showed that, in 2001, the NRC staff was well aware of the nature of the EPRI testing that the licensee relied on. The Panel did not find any evidence that the licensee claimed or the NRC staff believed that the valves were "qualified" in an ASME BPV Code certification sense; rather, the record shows that the NRC staff thoroughly considered the testing conducted on valves of the type installed at the plants and applied well-informed and reasoned technical judgment in reaching its conclusion that the EPRI testing provided appropriate qualification.

The Panel confirmed its conclusions and understanding about the 2001 NRC staff review via discussions with the individual who was the responsible Section Chief in the Reactor Systems Branch at the time. He informed the Panel that the 2001 license amendment was based on the exercise of staff engineering judgment and that there was no discussion of ASME BPV Code certification or qualification of valves. In addition, the Panel found that the NRC approved power uprates for other nuclear power plants that included comparable staff evaluations of water discharge through PORVs or PSVs based on test information provided by individual licensees. For example, in 2001, the NRC granted a power uprate for Shearon Harris that included the operability of PORVs and PSVs during the discharge of subcooled water, referencing TMI Action Plan Item II.D.1.[53] As noted above, in 2006, the NRC also granted a power uprate for

[52] NRC 1996

Beaver Valley. The SE for this Beaver Valley amendment referred to RIS 2005-29 and indicated that there was reasonable assurance that the PSVs would adequately discharge water and reseat following a spurious safety injection actuation, based on the EPRI test data from 1981 and an evaluation of the temperature of the liquid being discharged.

During the NRC evaluations of license amendments since the TMI-2 accident, the NRC staff has specified in some SEs that a PORV or PSV would be assumed to stick open if it was not qualified for liquid service. To address this concern, the NRC staff reviewed and accepted a variety of test information (including EPRI, Wyle, and vendor testing) submitted by individual licensees to demonstrate the capability of PORVs or PSVs to reseat following water discharge. In the sample of SEs it reviewed, the Panel did not find a specific requirement for the PORVs or PSVs to be certified under the ASME BPV Code as capable of ~~passing water~~reclosing after water discharge ~~and reclosing~~.

Comment [CT]: Edited slightly from Tom's change.

In 2004, the NRC issued license amendments for Byron and Braidwood granting an adjustment to the PSV setpoints. In an RAI, the NRC staff requested that the licensee perform a quantitative analysis regarding the number of opening cycles during which the PSV would be expected to pass water and the temperature of the water being discharged. In the Setpoint SE, the NRC staff concluded that the analysis was acceptable for assuring that the PSVs would remain operable following a spurious safety injection event.

3.9 RIS 2005-29 (2005) and Proposed Draft Revision 1 to RIS 2005-29 (2015)

In 2005, the NRC staff issued RIS 2005-29 "to notify licensees of a concern identified during recent reviews of power uprate [LARs]." The RIS addressed the manner in which some licensees acted in response to NSAL-93-013. The RIS was issued at the division level in NRR and does not include a record of office-level concurrence. The RIS was not reviewed by CRGR. The Panel requested information on the basis for the CRGR's decision not to review the proposed RIS before it was issued, but the CRGR staff could not find any related documentation. It appears to the Panel that the CRGR may not have reviewed the RIS because of assertions in the RIS such as these:

- "This RIS requires no action or written response and, therefore, is not a backfit under 10 CFR 50.109. Consequently, the NRC staff did not perform a backfit analysis."
- "This RIS is informational and pertains to a NRC staff position that does not depart from current regulatory requirements and practice."

A key statement in RIS 2005-29 is the following (with emphasis added):

> The NRC staff's position is noted in the power uprate review standard, as follows: "For the [IOECCS] and [CVCS] malfunctions that increase reactor coolant inventory events: (a) non-safety-grade pressure-operated relief valves should not be credited for event mitigation and (b) pressurizer level should not be allowed to reach a pressurizer water-solid condition.".

However, the NRC staff review standard cited in the RIS (RS-001) is explicitly limited to EPU reviews, stating that "[t]he staff does not intend to impose the criteria and/or guidance in this

[53] NRC 2001d

review standard on plants whose design bases do not include these criteria and/or guidance. No backfitting is intended or approved in connection with the issuance of this review standard."[54]

This intent of RS-001 to define and clarify the scope of EPU reviews, but not impose new requirements or new interpretations of requirements, was confirmed by the Panel in discussions with the manager responsible for developing and issuing RS-001. Therefore, contrary to the RIS statement, neither RS-001 nor RIS 2005-29 documented "known and established standards of the Commission" applicable to Byron and Braidwood.

The Panel also notes that neither RIS 2005-29 nor its draft Revision 1,[55] which is currently under development, discuss water discharge certification requirements in accordance with the ASME BPV Code. In fact, as stated above, the NRC issued a 2006 power uprate amendment for Beaver Valley in which the SE cited RIS 2005-29 and yet relied on the EPRI testing data to address the concern.

3.10 SECY-05-0138 (2005)

SECY-05-0138 presents a comprehensive history of the application of the single failure criterion, including extensive discussion of the treatment of passive components in fluid systems.[56] The paper enclosed a July 2005 draft of an NRC staff technical report on the single failure criterion. Section 4.2.2 of this report acknowledges that "[o]ne particular issue identified in this project is the continued existence of the footnote to the definition of single failure in 10 CFR [Part] 50 Appendix A stating that the regulatory position on considering passive failures in fluid systems is under development." In Section 2.5.3, the draft report quotes from SECY-77-439 (discussed above) and recognizes that in current practice, as in 1977, "[p]assive failures in fluid systems are generally excluded from single-failure assessments."

SECY-05-0138 and the accompanying draft report present three alternatives for using a risk-informed and performance-based approach to address the single failure issue. The draft report clarifies that all of the alternatives "could include developing a position on single passive failures in fluid systems to replace the footnote now in 10 CFR Part 50 Appendix A definitions."

These documents make it clear that, with few exceptions, neither the NRC staff nor the Commission has established specific requirements relating to the treatment of passive component failures in fluid systems. The Panel believes the existence of this Commission paper, contemporaneous with discussions on potential PSV failures (e.g., RIS 2005-29), makes it clear that no specific "known and established standards" on PSV failures had been developed between 1977 and the time of the Byron and Braidwood license amendments in 2001 and 2004.

3.11 Standard Review Plan Revision (2007)

Revision 2 to SRP Sections 15.5.1 – 15.5.2 states:

> If the plant is equipped with PORVs that are (1) safety-related equipment and (2) qualified for water relief, then they may be assumed to reseat properly after having relieved water. The [PSVs], too, may be assumed to reseat properly after having relieved water; but only if such valves have been qualified for water relief.

[54] NRC 2003
[55] NRC 2015a
[56] NRC 2005a

However, this section does not reference ASME BPV Code requirements for safety valve certification.

3.12 Backfit Letter and Subsequent Backfit Appeals (2015-2016)

The Backfit SE is predicated on the following positions:

- "water relief through a valve that is not qualified for water relief will cause that valve to stick in its fully open position" (emphasis added)
- "the licensee ... has not applied the single-failure assumption" (emphasis added)
- "nor [has the licensee] provided ASME water qualification documentation for the PSVs ... the ASME ... original Overpressure Protection Report ... inservice test history ... including both water and steam tests" (emphasis added)

The Backfit SE contends that an IOECCS would escalate to a more severe event. Such an escalation would be contrary to the Byron and Braidwood licensing basis (i.e., contrary to the ANS non-escalation position) and could be in non-compliance with the GDCs (as included in the Byron and Braidwood licensing basis) since an IOECCS with a stuck-open valve had not been analyzed and shown to meet the appropriate criteria for an AOO.

Based on its review of all the relevant documents and discussions with the individuals (staff and managers) involved in the original review and the backfit, the Panel has developed an understanding of the regulatory requirements and practices, the potential safety issues, and backfit rule obligations. The Panel has determined that the numerous, complex, and detailed regulatory and technical issues all depend on the answers to two critical questions on valve performance:

- Must the PSVs in question be assumed to fail given liquid water discharge because of the lack of ASME BPV Code certification for water discharge?
- Must the PSVs be assumed to fail in accordance with the GDC "single failure" requirements?

In the Backfit SE, the NRC staff indicated that "[o]ne assumption that is particularly important to the non-escalation criteria is that water relief through a valve that is not qualified for water relief will cause that valve to stick in its fully open position" (emphasis added). The Panel concluded that this issue—the treatment of potential valve failure—is not only "particularly important," it is the critical issue upon which the compliance backfit hinges.

Based on the historical evidence, the Panel concluded that there is not now, nor has there been, a known and established Commission standard (1) that PSVs must be assumed to fail following water discharge in the absence of ASME BPV Code certification for water discharge, or (2) that PSVs must be assumed to fail as part of single failure criterion analysis. The NRC staff's determination that ASME BPV Code certification is necessary first appears in the Backfit SE. The determination that application of the single failure criterion is necessary first appears in the draft Revision 1 to RIS 2005-29. The Panel has not identified these positions being stated in any final NRC requirement or guidance document.

The Panel also concluded that in 2001 and 2004 and at present, the known and established standard of the Commission is 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. In preparing the Uprate SE and the Setpoint SE, the NRC staff exercised reasonable and well-informed engineering judgment when the NRC staff concluded that the PSVs were unlikely to stick open. On the bases of its document reviews and interviews, the Panel concluded that the NRC staff reviewers involved in the 2001 power uprate review were among the most experienced and senior reviewers in their areas of expertise. The NRC staff valve expert involved in the review was the agency's most knowledgeable individual on PSVs and the relevant ASME Code requirements, and was a nationally recognized expert. The Panel did not find any evidence that the NRC staff's issuance of the 2001 or 2004 license amendments was based on an omission or mistake of fact. Rather, the Panel concluded that the current NRC staff positions on valve qualification in the Backfit SE are new or modified interpretations of compliance.

In interactions with the Panel, NRR staff emphasized several issues raised in the Backfit Letter. The Panel summarizes its consideration of those issues in the following subsections.

3.12.1 Non-Escalation Position and Valve Failure

In the Backfit SE, the NRC staff discussed the definition of event conditions in ANS-51.1/N18.2-1973 and the provision in this standard that events of one condition do not propagate to cause a more serious fault. This position is commonly known as the non-escalation position. In interactions with the Panel, NRR staff provided several clarifications on this topic, summarized by the Panel as follows:

- ANS-51.1/N18.2-1973 defines the categories of design basis transients and accidents based on an anticipated frequency of occurrence (annually for ANS Condition II events).
- It is a long-standing NRC position that escalation from one condition to another is not acceptable.
- ANS-51.1/N18.2-1973 constitutes a known and established standard that has been reflected in NRC guidance documents and in the licensing basis of each U.S. nuclear power plant.

The Panel confirmed that this ANS standard is referenced in several places in Chapter 15 of the Byron and Braidwood UFSAR. The Panel agrees that the non-escalation position is an established standard applicable to Byron and Braidwood, but did not identify historical evidence that implementation of this standard requires Exelon to assume that its pressurizer valves will fail open under water discharge conditions, to apply the single failure criterion to PSV failure in these circumstances, or to impose ASME Code requirements for certification, qualification, or testing of PSVs for water discharge.

3.12.2 Non-Escalation Position and Return to Service

In the Backfit SE, the NRC staff makes reference to the time it would take to clean up a contaminated containment following a stuck-open pressurizer valve. In interactions with the Panel, NRR staff re-emphasized concerns that extended steam and water discharge through the pressurizer valves would result in the failure of the pressurizer relief tank rupture disk, would require repair of the damaged PSVs, and might cause an extended time period for the return to service of the nuclear power plant.

The Panel does not consider the time period necessary for the licensee to perform radioactive clean-up activities in the containment building, to inspect and conduct any necessary repairs to the PSVs, or to prepare for plant startup, to constitute issues that support a compliance backfit imposed by the NRC. The NRC staff would verify (e.g., through inspection) that the licensee ~~would~~ had conducted these activities appropriately to protect the public health and safety prior to plant restart. The Backfit SE states that UFSAR Section 15.5.1.3 "implie[s]" that the plant will return to operation in a "short period," but the Panel found no bases for a timing requirement in UFSAR Section 15.5.1.3. Also, the Panel did not find a regulatory requirement or basis for defining or limiting the time available for the plant to return to operation.

3.12.3 TMI Action Plan Item II.D.1 and EPRI Testing

Although the Backfit Letter and NRR Appeal Decision do not speak explicitly to TMI Action Plan Item II.D.I, in interactions with the Panel, NRR staff stated that the known and established standard in question is the TMI Action Plan Item II.D.I standard for licensees and applicants to conduct testing to qualify the RCS relief and safety valves under expected operating conditions for design-basis transients and accidents. As discussed above and in Appendix B to this report, the NRC accepted the EPRI testing to satisfy TMI Action Plan Item II.D.1 for Byron and Braidwood in SEs forwarded by letters in 1988 and 1990. Therefore, the Panel concludes that this known and established standard referenced by the NRC staff had been met for Byron and Braidwood.

In interactions with the Panel, the NRR staff further stated that an omission or mistake of fact occurred when the licensee failed to acknowledge that the EPRI testing program did not evaluate water discharge from the pressurizer valves during extended high pressure safety injection for Byron and Braidwood. As discussed in Appendix B to this report, in the 1988 and 1990 SEs for the Byron and Braidwood responses to TMI Action Plan Item II.D.1, the NRC staff evaluated the capability of the PSVs and PORVs during feedwater line break accidents, including water discharge. In these SEs, the NRC staff found that the performance of the PSVs and PORVs with water discharge was acceptable based on the EPRI tests. Therefore, the Panel also concluded that the licensee's reference to the EPRI testing program was not an omission or a mistake of fact.

3.12.4 ASME Code Certification

In the Backfit SE, the NRC staff stated that certain ASME Code information would be necessary to support water qualification of the PSVs. In interactions with the Panel, NRR staff stated that, to satisfy the standard for water discharge capability of pressurizer valves, it would be necessary to conduct flow capacity certification in accordance with the ASME BPV Code and inservice testing throughout the service life in accordance with the ASME OM Code. The NRR staff referenced certain licensing actions in which water discharge was not considered acceptable, or different actions were required.[57]

As discussed in Appendix C to this report, the NRC staff required additional actions for some licensees to support reliance on the PORVs for water discharge and to avoid water discharge through the PSVs. The Panel found, however, that the NRC staff also allowed some licensees to rely only on EPRI testing without significant additional activities. The Panel did not identify instances where the NRC staff imposed certification by the ASME BPV Code and testing in

[57] Salem (NRC 1997), Millstone (NRC 1998), and Callaway (NRC 2000)

accordance with the OM Code, or required alternatives to the ASME BPV or OM Codes, in the examples of NRC staff review of water discharge capability for pressurizer valves.

The NRR staff also identified for the Panel specific ASME Code provisions that it viewed as supporting its position that ASME Code requirements apply to qualification of pressurizer valves for water discharge. The NRR staff, however, did not provide evidence that the NRC staff has consistently interpreted these provisions as the NRC staff is now interpreting them. Given the NRC staff's resolution of TMI Action Plan Item II.D.1 and the variations in the NRC staff's licensing practices, the Panel concludes that the NRR staff's current application of the ASME Code is not supported by the historical record.

3.12.5 Conduct of 2001 and 2004 License Amendment Reviews

In light of the wide range of positions taken by the NRC staff during its reviews of pressurizer valve capability since the TMI-2 accident, the Panel agrees that, in the course of preparing the 2001 Uprate SE or Setpoint SE, the NRC staff could have considered the need for the licensee for Byron and Braidwood to improve the reliability of the PSVs or PORVs for water discharge or to avoid water discharge through the PSVs by PORV improvements. The NRC staff may have been able to justify additional actions, but they determined that it was not necessary. Instead, the NRC staff reviewers in 2001 used their expert engineering judgement to determine that it was not necessary to assume that the PSVs or PORVs would stick open with water discharge, based on EPRI test information, licensee supplemental information, and their own technical experience.

In discussions with the Panel, NRR staff raised a concern that the Setpoint SE does not document a re-review of the qualification of the PSVs and noted that if the Uprate SE had not found water discharge through the PSVs to be acceptable, it is unlikely that the NRC staff would have approved this 2004 amendment. In Appendix C to this report, the Panel summarizes the discussion in the Setpoint SE of the PSV water discharge capability. The Panel recognizes that a staff review may rely on a previous more extensive review to determine the acceptability of a similar request. The Panel does not consider the review approach used in 2004 to challenge the acceptability of the 2001 review.

4 RESPONSE TO THE EDO QUESTIONS

In establishing the Panel, the EDO asked the Panel to answer five specific questions, as well as evaluating the overall appropriateness of the backfit. The Panel's answers to these questions are provided below.

4.1 Were the approvals based on a mistake? If so, what was the mistake and what are the implications for Braidwood and Byron?

In responding the question, the Panel has considered the differing views of the NRR staff and the licensee on this issue. Those positions are summarized below:

- In the NRR Appeal Decision, the NRC staff claims that "[t]he NRC erred in approving a sequence of events that allowed the [IOECCS], [CVCS] malfunction, and inadvertent opening of a pressurizer safety or relief valve analyses in the 2001 and 2004 [SEs]" and "the NRC staff understood the PSVs to be qualified for water relief when, in fact, they were not."

- Exelon claims in the NRR Backfit Appeal that "the compliance exception requires more than simply asserting that the prior staff approvals were wrong—the NRC must demonstrate that the prior approvals were erroneous because of an omission or mistake of fact at the time of the approval. The NRC has not made that case here."

On the basis of its independent review, the Panel concluded that, in 2001 and 2004, the NRC staff did not misunderstand the qualification status of the PSVs and that it was not a mistake to undertake a review of or make a technically based safety finding on the likely successful performance of the valves. In the Panel's opinion, the actions of the Reactor Systems Branch in 2001 to reach out to the Division of Engineering's Mechanical Engineering Branch for expert technical review assistance was both appropriate and commendable. After considering the materials presented by the licensee in support of the 2001 and 2004 requests and discussing the 2001 review with one of the involved managers, the Panel found no indication that the senior reviewer evaluating the topic was misled regarding the qualification status of the PSVs, but rather used his expert judgment in determining the appropriate level of qualification for a technically complex topic for which there was not a single accepted approach. For these reasons, the Panel concluded that the NRC staff reviews and approvals of the 2001 and 2004 license amendments were not based on omissions or mistakes of fact.

4.2 What is the known and established standard for water qualification of PSVs?

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.

4.3 What is the known and established standard for progression of postulated events between categories of severity?

For Byron and Braidwood, the NRC staff and the Panel agreed that the known and established standard for progression of postulated events between categories of severity is the "non-escalation position" specified in ANS-51.1/N18.2-1973. This position, which is included in the Byron and Braidwood UFSAR, requires that events of one condition do not propagate to cause a more serious condition (i.e., from ANS Condition II to ANS Condition III or IV). The Panel concluded that the IOECCS (an AOO per the GDC definition and an ANS Condition II event) would escalate to a more severe event <u>if</u> a PSV were to stick open, or <u>if both</u> a PORV stuck open and its block valve failed to close. Such an escalation would be contrary to the Byron and Braidwood licensing basis (i.e., contrary to the ANS non-escalation position) and could be in non-compliance with the GDC (as included in the Byron and Braidwood licensing basis), since an IOECCS with a stuck-open valve had not been analyzed and shown to meet the appropriate criteria for an AOO. However, this event progression standard does not establish specific standards for valve qualification to determine whether a valve would stick open and cause this escalation. Therefore, the Panel concluded that it is not the basis for a compliance backfit given the current set of facts. (Additional information about ANS-51.1/N18.2-1973 is included in Section 3.12.1 of this report.)

4.4 Does the current licensing basis for Braidwood and Byron comply with the applicable regulations? Is it adequate to provide protection to public health and safety?

The Panel concluded that the current licensing basis for Byron and Braidwood complies with the applicable regulations based on the UFSAR analyses, which the NRC staff found acceptable through a reasonable and technically sound evaluation using appropriate Commission safety standards. This licensing basis has been determined by the NRC staff to provide adequate protection to public health and safety.

4.5 Given that Exelon suggests that the NRC pursue a cost-justified substantial safety enhancement backfit, what is the contribution to overall plant risk of the current configuration at Braidwood and Byron?

The Panel requested RES to provide information and insights on the risk significance of the sequence at issue, to assure that the Panel's judgments were being made with a full understanding of their significance, and to assist in responding to the EDO question.

The RES study[58] suggests that the most significant IOECCS sequence, assuming that all pressurizer overfill events lead to a small LOCA, contributes approximately 1 percent of the total internal event core damage frequency (CDF). In its report, RES estimated that the maximum benefit (CDF reduction) of 1.5E-07 per year would be achieved if the plants were modified (backfit) such that pressurizer overfilling was always prevented. If the PSVs are not assumed to always fail following water discharge (consistent with the NRC staff expert judgment in 2001) or if the plants were modified in a different way that did not ~~ensure~~ prevent pressurizer overfilling, the risk-reduction benefit of implementing the backfit would be even smaller.

The Panel is aware of and sensitive to two important issues related to this question. First, NRR, not the Panel, is responsible for any decisions on alternative application of the backfit rule to this issue (through the other categories of adequate protection or cost-justified substantial safety enhancement). Second, the Panel does not wish to imply that "the contribution to plant risk" should be seen as the only measure of enhanced safety. The issues of event classification and the non-escalation of events are essentially defense-in-depth concepts. Defense in depth has a recognized role and value in the regulatory process. The Panel is also aware that not every defense-in-depth feature has the same safety significance, and that the estimated risk significance (measured in core damage frequency) is very relevant.

Within the context described above, the Panel concluded that the contribution to overall plant risk is very small.

5 SUMMARY AND CONCLUSIONS

The compliance exception to the Backfit Rule is intended to address failures to meet known and established Commission standards because of omission or mistake of fact. New or modified interpretations of what constitutes compliance do not fall within the exception. Therefore, to address the appeal of the proposed compliance backfit, the Panel focused on determining if this case is most appropriately characterized as one in which the licensee "failed to meet known and

[58] NRC 2016f

established standards of the Commission because of omission or mistake of fact," or rather as a case of a "new or modified interpretations of what constitutes compliance."

The NRC staff's compliance backfit argument depends on two separate determinations:

1. the assumed failure of PSVs to reclose after passing water, and

2. the necessity of preventing "event escalation" (i.e., the position that "an incident of moderate frequency should not generate a more serious plant condition without other faults occurring independently").

For the NRC staff's compliance backfit conclusion to be valid, both of these determinations must meet the above compliance backfit standard by involving failure to meet known and established standards of the Commission.

In the first of these determinations, the NRC staff's compliance backfit is based on the assumption in the Backfit SE that the PSV fails to reclose given the absence of "ASME water qualification documentation." As indicated in the Backfit SE, the Uprate SE involved a technical evaluation of safety valve capability and likely performance under water-discharge conditions rather than a simple assumption of a failure. The NRR Appeal Decision indicates that "the 2001 and 2004 [license amendment] approvals occurred because the NRC staff understood the PSVs to be qualified for water relief when, in fact, they were not."

The Panel carefully considered these views and has reviewed the relevant documents including the licensee's responses to the NRC staff's RAIs,[59] the NRR technical branch's SE input,[60] and the Uprate SE. 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 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.

On the basis of its independent review, the Panel concluded that the NRC staff who prepared the Uprate SE did not misunderstand the qualification status of the PSVs and that it was not a mistake to undertake a review of or make a technically based safety finding on the likely successful performance of the valves. In the Panel's opinion, the actions of the Reactor Systems Branch in 2001 to reach out to the Division of Engineering's Mechanical Engineering Branch for expert technical review assistance was both appropriate and commendable. After considering the materials presented by the licensee in support of the requests and discussing the review with one of the involved managers, the Panel found no indication that the senior reviewer evaluating the topic in 2001 was misled regarding the qualification status of the PSVs, but rather used his expert judgment in determining the appropriate level of qualification for a technically complex topic for which there was not a single accepted approach. For these reasons, the Panel concluded that the NRC staff review documented in the Uprate SE was not ~~not~~ based on omissions or mistakes of fact.

The Panel concluded that three related technical and regulatory positions related to the PSVs (separate from the issue of the non-escalation position) underpin the backfit:

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[59] ComEd 2000b, Exelon 2001

[60] NRC 2001a

1. ASME water qualification (certification) documentation is required if a valve is to be assumed to reclose after passing water.

2. Water discharge through a steam-qualified valve will cause that valve to stick in its fully open position.

3. PSVs are subject to a single-failure assumption.

In the Panel's view, none of these three positions were "known and established standards of the Commission" in 2001 or 2004 for determining when it was appropriate to assume a failure of PSVs to reseat. In fact, they were not "known and established standards of the Commission" in 2005 (when RIS 2005-29 was issued) or 2006 (when the Beaver Valley EPU was approved) or 2007 (when Revision 2 to SRP Sections 15.5.1 – 15.5.2 was issued).

Moreover, these positions do not appear to be "established standards of the Commission" at present. The 2007 version of SRP Sections 15.5.1 – 15.5.2 allows credit for PORVs and PSVs if they have been "qualified for water relief." The NRC staff's determination that ASME BPV Code certification is necessary first appears in the Backfit SE and is not addressed in any of the final NRC requirements or guidance documents reviewed by the Panel. The determination that application of the single failure criterion is necessary first appears in the draft Revision 1 to RIS 2005-29, which is still under development, and is not included in any final NRC requirement or guidance document reviewed by the panel.

The Panel 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. In earlier documents addressing this topic, beginning with NUREG-0737, it is the Panel's view that the use of the word "qualified" or "qualification" implies a general demonstration of capability, such as in the EPRI testing done in response to TMI Action Plan Item II.D.1. In light of this standard, the Panel concluded that, when preparing the Uprate SE and the Setpoint SE, the NRC staff exercised reasonable and well-informed engineering judgment to conclude that the PSVs were unlikely to stick open.

Overall, the Panel concluded that the NRC staff's position on valve qualification in the Backfit SE is a new or modified interpretation of what constitutes compliance in addressing potential PSV failures following water discharge. Although this new staff position represents a well-intentioned and conservative approach that could provide additional safety margin, the Panel concluded that it does not provide a basis for a compliance backfit.

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.

The Panel's findings, therefore, support the Exelon backfit appeal.

6 ADDITIONAL PANEL THOUGHTS

In addition to the specific finding relating to the backfit appeal, the Panel believes it is important to acknowledge, and for the NRC staff and licensees to appreciate, that water discharge through a PSV not specifically designed for such service is undesirable and should be minimized or avoided as a matter of conservative engineering and prudent operations. This is reinforced by the information provided in NSAL-93-013 and its Supplement 1, and the actions by various licensees in response to these documents, as well as the limited scope of the EPRI testing conducted over 30 years ago.

Operator training, control room procedures to terminate the event before pressurizer filling, and use of PORVs rather than reliance on PSVs, are clearly preferred and prudent measures, whether they form the facilities' UFSAR licensing basis and are assumed in the accident analyses or not.

The PSVs in question were designed for steam service. Steam relief is their normal service condition and applies to their ASME BPV Code certification. The Panel supports the previous NRC staff determinations for Byron and Braidwood and certain other plants that PSVs experiencing water discharge during an abnormal or accident condition need not be assumed to fail since there was a reasonable and technically well-informed engineering judgement to the contrary. However, the Panel also considers the actions by various licensees to improve the reliability and performance of the PORVs to avoid water discharge through the PSVs to be prudent in light of the design specifications of the PSVs.

The Panel considered but could not determine the extent to which the licensee for Byron and Braidwood addressed crediting water discharge through the PSVs, PORVs, or PORV block valves in the Byron and Braidwood inservice testing programs. The Panel recognizes that the difference between the intended use of these valves for overpressure protection and their infrequent use in response to certain plant events might be considered in implementing appropriate inservice testing activities.

The Panel notes that water discharge through various pressurizer valves is not a new issue because water discharge has always been credited (by the licensee for Byron and Braidwood and other licensees) for the feedwater line break analysis in UFSAR Section 15.2.8.

On the basis of its review, the Panel also noted that the issue of pressurizer valve performance following water discharge appears to have generic applicability, and is not specific to only Byron and Braidwood. The Panel believes that resolution of this issue would have benefited from consideration of the generic nature of the issue through the appropriate NRC processes. The Panel included the information it gathered and assessed to reach its conclusion regarding the generic nature of the issue in Appendices B and C of this report. Should the NRC staff undertake a generic look of the issues, it should, among other things, consider the information presented and questions raised in those appendices. The review should also include a reassessment of the information and staff positions communicated in RIS 2005-29, as well as those included in its proposed Revision 1, which is currently under development, to determine whether or not these documents include new staff positions with the potential for inappropriate or unintended backfitting. As part of any generic assessment, the Panel also recommends that staff determine whether the information in RIS 2005-29 and its proposed Revision 1 should be incorporated into a regulatory guide or another guidance document.

APPENDIX A: HISTORY OF THE BACKFIT RULE AND THE COMPLIANCE EXCEPTION

The Backfit Rule

Title 10 of the *Code of Federal Regulations* (10 CFR), Section 50.109, "Backfitting," was originally promulgated in 1970.[61] Because of perceived deficiencies in the rule, the U.S. Nuclear Regulatory Commission (NRC) substantially revised it in 1985.[62] The 1985 rule was challenged in court, and the U.S. Circuit Court for the District of Columbia (D.C. Circuit) vacated this rule in its entirety. The D.C. Circuit took this action because it concluded that the revised rule could be interpreted to allow the NRC to consider costs in defining or redefining what is required for adequate protection of the public health and safety.[63] In response, the NRC revised the Backfit Rule in 1988 to remove any implication that costs could be considered in defining or redefining adequate protection.[64] The 1988 revisions only differed from the 1985 rule to the extent necessary to address the court's concerns. The 1988 rule was also challenged in court, but this time the D.C. Circuit upheld the rule.[65]

In its current form, 10 CFR 50.109(a)(1) defines backfitting as

> … the modification of or addition to systems, structures, components, or design of a facility; or the design approval or manufacturing license for a facility; or the procedures or organization required to design, construct or operate a facility; any of which may result from a new or amended provision in the Commission's regulations or the imposition of a regulatory staff position interpreting the Commission's regulations that is either new or different from a previously applicable staff position … .

Unless one of three specified exceptions apply, the NRC may impose a backfit only if it performs a backfit analysis in accordance with 10 CFR 50.109(a)(2) and determines in accordance with 10 CFR 50.109(a)(3) "that there is a substantial increase in the overall protection of the public health and safety or the common defense and security to be derived from the backfit and that the direct and indirect costs of implementation for that facility are justified in view of this increased protection."

Section 50.109(a)(4) sets forth the three exceptions to the requirements of 10 CFR 50.109(a)(2) and (a)(3). The first exception, the compliance exception, applies if the "modification is necessary to bring a facility into compliance with a license or the rules or orders of the Commission, or into conformance with written commitments by the licensee." 10 CFR 50.109(a)(4)(i). The second and third exceptions relate to actions ensuring adequate protection or to actions that involve defining or redefining adequate protection. 10 CFR 50.109(a)(4)(ii)-(iii).

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[61] AEC 1970 (Author and year citations in footnotes refer to the designation of references in Appendix D to this report.)
[62] NRC 1985
[63] *Union of Concerned Scientists v. U.S. Nuclear Regulatory Com'n*, 824 F.2d 108, 119-20 (1987).
[64] NRC 1988b
[65] *Union of Concerned Scientists v. U.S. Nuclear Regulatory Com'n*, 880 F.2d 552 (1989).

Commission Policy

The Commission addressed its intended application of the compliance exception in the 1985 rulemaking:[66]

> The compliance exception is intended to address situations in which the licensee has failed to meet known and established standards of the Commission because of omission or mistake of fact. It should be noted that new or modified interpretations of what constitutes compliance would not fall within the exception and would require a backfit analysis and application of the standard.

In the 1985 rule, the Commission acknowledged that staff interpretations of regulations are not legally binding, but the Commission also stated that "staff interpretations of broadly stated rules are often necessary to give a rule effect and in some instances may be a causal factor in initiating a backfit."[67] The Commission also stated, "Many of the most important changes in plant design, construction, operation, organization, and training have been put in place at a level of detail that is expressed in staff guidance documents which interpret the intent of broad, generally worked [sic] regulations."[68]

Backfitting Guidance

Extensive information regarding the appropriate implementation of backfitting is provided in NUREG-1409.[69] Relevant excerpts from this guidance are provided below.

Applicable Regulatory Staff Positions

According to NUREG-1409, to be a backfit, "a new or revised staff position or requirement must be involved, that is, there must be a change in content or applicability of the previously applicable regulatory staff position (in the direction of increased safety requirements) … ." An applicable regulatory staff position is a requirement or position already specifically imposed on or committed to by a licensee. Examples of applicable regulatory staff positions include:

- legal requirements, as in explicit regulations, orders, and plant licenses and in amendments, conditions, and technical specifications

- written licensee commitments such as those contained in the final safety analysis report, licensee event reports, and docketed correspondence, including responses to NRC bulletins, generic letters, inspection reports, or notices of violation and confirmatory action letters

- NRC staff positions that are documented explicit interpretations of more general regulations and are contained in documents such as the Standard Review Plan, branch technical positions, regulatory guides, generic letters, and bulletins

A similar list of examples is provided in Manual Chapter 0514,[70] which is also included as Appendix D to NUREG-1409. Manual Chapter 0514 was referenced in the 1988 rulemaking,

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[66] NRC 1985, at 38103
[67] *Id.* at 38102
[68] *Id.* at 38103. The 1988 rulemaking neither revised the compliance exception as stated in the 1985 rule nor provided additional guidance on its interpretation.
[69] NRC 1990c

and a working draft was provided to the Commission for information in SECY-88-102.[71] Manual Chapter 0514 provides a definition of "applicable regulatory staff positions" that is slightly more detailed than the definition in NUREG-1409. This definition from Manual Chapter 0514 is quoted below, with additional detail beyond NUREG-1409 emphasized in underlined text.

> Applicable regulatory staff positions are those already specifically imposed upon or committed to by a licensee at the time of the identification of a plant-specific backfit, and are of several different types and sources:
>
> a. Legal requirements such as in explicit regulations, orders, plant licenses (amendments, conditions, technical specifications). Note that some regulations have update features built in, as for example, 10 CFR 50.55a, Codes and Standards. Such update requirements are applicable as described in the regulation.
>
> b. Written commitments such as contained in the [Final Safety Analysis Report], [Licensee Event Reports], and docketed correspondence, including responses to Bulletins, responses to Generic Letters, Confirmatory Action Letters, responses to Inspection Reports, or responses to Notices of Violation.
>
> c. NRC staff positions[72] that are documented, approved, explicit interpretations of the more general regulations, and are contained in documents such as the [Standard Review Plan], Branch Technical Positions, Regulatory Guides, Generic Letters, and Bulletins; and to which a licensee or an applicant has previously committed to or relied upon. Positions contained in these documents are not considered applicable staff positions to the extent that staff has, in a previous licensing or inspection action, tacitly or explicitly excepted the licensee from part or all of the position.[73]

How Regulatory Positions are Established

NUREG-1409 provides responses to a number of questions regarding backfitting. The following response was given to questions asking, "Is it appropriate for the NRC staff to rely on informal or formal communications to other licensees as official NRC positions? What about NRC tacit approval of documents?"

> Informal or formal communications to one licensee are not official positions to all licensees. Section 053 of Manual Chapter 0514 identifies what can be applied as official staff positions in a plant-specific context. They are legal requirements such as contained in explicit regulations, orders, and plant licenses; written commitments such as contained in final safety analysis reports, licenses event reports, and docketed correspondence; and documented, approved explicit interpretations such as contained in the [Standard Review Plan], branch technical positions, regulatory guides, generic letters, and bulletins. Orders, licenses, and written commitments are applicable only to a particular licensee.

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[70] NRC 1988c

[71] NRC 1988a

[72] Requirements may be imposed by rule or order. Staff interpretations such as examples of acceptable ways to meet requirements are not requirements in and of themselves.

[73] Imposition of a staff position from which a licensee has previously been excepted is a backfit.

> If the NRC staff previously exempted a licensee from a legal requirement or approved position, it is not applicable to that licensee for the purpose of backfit consideration. Explicit exemption would be done formally in writing. The Appendix to NRC Manual Chapter 0514 discusses tacit approval under reanalysis of issues. Two situations are covered. In the first case, staff review of a previously accepted licensee action or program may result in a requested change. This would be classified as a backfit because it represents a change in a previous staff position and would require a backfit analysis (or a documented evaluation if it meets one of the exceptions listed in the backfit rule). In the second case, a licensee submittal committing to a specific course of action that has not received timely NRC staff review is implemented by the licensee. In this case, it is considered that the NRC staff tacitly accepted the licensee's action since timely notice to the contrary was not given. If the NRC staff subsequently adopts a different position and requests a change in the licensee action, this change may be classified as a backfit and thus require a backfit analysis (or a documented evaluation if it meets one of the exceptions listed in the backfit rule).

NUREG-1409 also addresses a question regarding tacit approvals by an inspector: "If an inspector has previously accepted (i.e., provided tacit approval of) a licensee's method, does a specific request for change constitute a backfit and if so, is a backfit analysis required?" The response is:

> Cases where an inspector provides tacit approval are relatively rare. Simply not challenging a licensee's practice normally would not be considered tacit approval. The only example provided in Manual Chapter 0514 is a case where the NRC has indicated tacit approval by not acting in a reasonable time on a licensee submittal and the licensee has moved ahead to implement the proposal described in the submittal. For the purpose of this question, it would most likely arise in connection with review of a licensee response to an inspection report.
>
> Explicit approval could be provided in an inspection report that states that a particular approach is acceptable. However, conclusions of that nature are usually made in [safety evaluations] rather than inspection reports.

Compliance Backfit Guidance

NUREG-1409 gives the following response to the question, "[h]ow does the backfit rule apply to new staff positions that reflect an evolving understanding of technical issues?"

> An evolving understanding of issues does not, by itself, define which category fits a particular backfit. Judgment must be applied to the facts of each particular case to determine whether the backfit is for compliance, to provide adequate protection, to redefine adequate protection, or to achieve a cost-justified substantial safety enhancement. For example, with regard to compliance, the 1985 statement of considerations for 10 CFR 50.109 indicates that "the compliance exception is intended to address situations where the licensee has failed to meet known and established standards of the Commission because of omission or mistake of fact....new or modified interpretations of what constitutes compliance would not fall within the exception...."

NUREG-1409 also provides an example where an evolving understanding of technical issues resulted in a compliance backfit that was apparently justified for at least some licensees. In

response to industry claims that Bulletin 88-11[74] lacked any backfitting justification, the NRC staff responded:

> Although the justification was not printed in the bulletin, NRC Bulletin 88-11, "Pressurizer Surge Line Thermal Stratification," was justified as a backfit. It is an example of a backfit that was determined by the responsible NRC official to be required as a matter of compliance with existing requirements and commitments. The CRGR reviewed the bulletin and concurred. The regulations currently require licensees to meet the applicable codes of the American Society of Mechanical Engineers (ASME), *Boiler and Pressure Vessel Code*. Because of the NRC staff's concern with the integrity of the surge line, licensees were requested to perform their fatigue analysis in accordance with the latest ASME Section III requirements that incorporate high cycle fatigue analysis. The justification provided by the NRC staff was that previously unconsidered thermal stratification phenomenon may invalidate the existing analysis performed to confirm the integrity of the surge line.
>
> Subsequently, it was understood that some licensees believed that the NRC staff's rationale was in error because they were not committed to the latest ASME Section III requirements by virtue of their license commitment. However, the issue became moot because these licensees undertook the analysis voluntarily in view of the safety importance of the issue and the fact that previous versions of the ASME Code did not completely address the concern.

[74] NRC 1988e

APPENDIX B: QUALIFICATION OF PRESSURE RELIEF VALVES IN NUCLEAR POWER PLANTS IN RESPONSE TO THE TMI-2 ACCIDENT

Byron and Braidwood Design and Code Requirements

Nuclear power plants in the United States use various types of pressure relief valves to protect personnel and equipment from overpressure events within reactor fluid systems. Pressure relief valves include safety valves, safety relief valves, and relief valves, with different designs, operating conditions, and requirements. The American Society of Mechanical Engineers (ASME) *Boiler and Pressure Vessel Code* (BPV Code), Section III, Division 1, specifies requirements for the design, operation, installation, and testing of pressure relief valves used for various functions in nuclear power plants.[75] For example, the ASME BPV Code (2007 Edition) in Article NB-7000, Overpressure Protection, specifies requirements several service conditions:

- steam and air or gas service for safety valves;
- steam, air or gas, and liquid service for safety relief valves;
- liquid service for relief valves; and
- steam, air or gas, and liquid service for pilot operated or power actuated pressure relief valves.

The ASME *Code for Operation and Maintenance of Nuclear Power Plants* (OM Code) provides requirements for the preservice and inservice testing (IST) programs for pressure relief valves in nuclear power plants.

Byron, Units 1 and 2 (Byron) and Braidwood, Units 1 and 2 (Braidwood) are Westinghouse-designed pressurized-water reactors (PWRs) that received their construction permits under Title 10 of the *Code of Federal Regulations* (10 CFR), Part 50, in December 1975. The pressurizer for each unit is equipped with three pressurizer safety valves (PSVs) and two power-operated relief valves (PORVs). The three PSVs are Crosby Model HP-BP-86, size 6M6 (6-inch), spring-loaded pop type, opened by direct fluid pressure. The PORVs are Copes-Vulcan Model D-100-160 3-inch pneumatic-actuated globe valves that respond to a signal from the pressure sensing system or to manual control. Each PORV can be isolated by a motor-operated block valve.

The ASME BPV Code of record for the design of the PSVs at Byron and Braidwood is the 1971 Edition through the Winter 1972 addenda of the ASME BPV Code, Section III. The ASME BPV Code applicable to Byron and Braidwood includes requirements for overpressure protection, including the following:

- Section NB-7300, "Overpressure Protection Report," in NB-7320(f) requires that the report include the redundancy and independence of the pressure-relief devices and their associated pressure-sensing and controls systems employed to preclude a loss of overpressure protection in the event of a failure of any pressure-relief device, or its sensing element, or its associated control, or an external power source.

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[75] References to individual ASME Code publications are not provided in Appendix D, but they are publicly available from ASME for a fee.

- Paragraph NB-7411, “Relieving Capacity of Pressure-Relief Devices,” specifies that the total rated relieving capacity shall be sufficient to prevent a rise in pressure of more than 10 percent above system design pressure (at design temperature) within the pressure-retaining boundary of the system, under any pressure transient anticipated to arise as summarized in the Overpressure Protection Report.

- Paragraph NB-7421, “Required Number and Capacity of Pressure-Relief Devices for Nuclear Systems,” states that the required relieving capacity intended for overpressure protection of a nuclear power system or portions of the system shall be secured by the use of at least two pressure-relief devices.

At the time of the Byron and Braidwood operating license review, Revision 1 of Standard Review Plan (SRP) Sections 15.5.1-15.5.2 and Section 15.6.1 provided general staff guidance for these plant transients.[76] In March 2007, the NRC staff issued Revision 2 to these SRP sections with significantly more detail, including a statement that PSVs and PORVs are assumed to fail open if they relieve water without being qualified.[77]

Actions Following Three Mile Island, Unit 2 Accident

The accident at Three Mile Island, Unit 2 (TMI-2) on March 28, 1979, included failure of a PORV on the pressurizer to reclose properly during the event. Based on lessons learned from the TMI-2 accident, the NRC issued recommendations regarding performance testing of safety and relief valves used in nuclear power plants in NUREG-0578.[78] In particular, the NRC staff recommended in Section 2.1.2, “Performance Testing for BWR [boiling-water reactor] and PWR Relief and Safety Valves,” of NUREG-0578 that nuclear power plant licensees commit to provide performance verification by full-scale prototypical testing for all relief and safety valves.

In October 1980, the NRC issued a letter to all then-operating nuclear power plants and applicants for operating licenses and holders of construction permits forwarding NUREG-0737.[79] TMI Action Plan Item II.D.1 in NUREG-0737 specified the NRC position that PWR and BWR licensees and applicants shall conduct testing to “qualify” the reactor coolant system (RCS) relief and safety valves under expected operating conditions for design-basis transients and accidents. The detailed clarification in NUREG-0737 of this NRC position specified the following:

> Licensees and applicants shall determine the expected valve operating conditions through the use of analyses of accidents and anticipated operational occurrences referenced in Regulatory Guide 1.70, Revision 2. The single failures applied to these analyses shall be chosen so that the dynamic forces on the safety and relief valves are maximized. Test-pressures shall be the highest predicted by conventional safety analysis procedures. [RCS] relief and safety valve qualification shall include qualification of associated control circuitry, piping, and supports, as well as the valves themselves.
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> A. Performance Testing of Relief and Safety Valves--The following information must be provided in report form by October 1, 1981:

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[76] NRC 1981b and NRC 1981c
[77] NRC 2007b and NRC 2007c
[78] NRC 1979a
[79] NRC 1980b and NRC 1980c

(1) Evidence supported by test of safety and relief valve functionability for expected operating and accident (non-[anticipated transient without scram]) conditions must be provided to NRC. The testing should demonstrate that the valves will open and reclose under the expected flow conditions.

(2) Since it is not planned to test all valves on all plants, each licensee must submit to NRC a correlation or other evidence to substantiate that the valves tested in the EPRI (Electric Power Research Institute) or other generic test program demonstrate the functionability of as-installed primary relief and safety valves. This correlation must show that the test conditions used are equivalent to expected operating and accident conditions as prescribed in the final safety analysis report (FSAR). The effect of as-built relief and safety valve discharge piping on valve operability must also be accounted for, if it is different from the generic test loop piping.

(3) Test data including criteria for success and failure of valves tested must be provided for NRC staff review and evaluation. These test data should include data that would permit plant-specific evaluation of discharge piping and supports that are not directly tested.

In describing the type of review to be conducted for this regulatory position, the NRC staff stated the following:

Pre-implementation review will be performed for EPRI and BWR test programs with respect to qualification of relief and safety valves. Also, the applicants' proposal for functional testing or qualification of PWR valves will be reviewed. Post-implementation review will also be performed of the test data and test results as applied to plant-specific situations.

In specifying the documentation required to satisfy this regulatory position, the NRC staff stated the following:

Pre-implementation review will be based on EPRI, BWR, and applicant submittals with regard to the various test programs. These submittals should be made on a timely basis as noted below, to allow for adequate review and to ensure that the following valve qualification dates can be met:

Final PWR (EPRI) Test Program--July 1, 1980

Final BWR Test Program--October 1, 1980

Block Valve Qualification Program--January 1, 1981

Post-implementation review will be based on the applicants' plant-specific submittals for qualification of safety relief valves and block valves. To properly evaluate these plant-specific applications, the test data and results of the various programs will also be required by the following dates:

PWR (EPRI)/BWR Generic Test Program Results--July 1, 1981

Plant-specific submittals confirming adequacy of safety and relief valves based on licensee/applicant preliminary review of generic test program results--July 1, 1981

Plant-specific reports for safety and relief valve qualification--October 1, 1981

Plant-specific submittals for piping and support evaluations--January 1, 1982

Plant-specific submittals for block valve qualification--July 1, 1982

EPRI Testing

In October 1982, EPRI issued NP-2670-LD to address testing of PORVs.[80] This report has been referenced by certain licensees (e.g., Section 15.2.14 of the North Anna, Units 1 and 2 Updated Final Safety Analysis Report (UFSAR)[81]).

In December 1982, EPRI issued NP-2628-SR, which described safety and relief valve tests for types of valves in service at nuclear power plants.[82] In particular, Section 3.5 documented the testing of Crosby safety valves similar to the PSVs at Byron and Braidwood, including two water tests. The report indicated chattering of the safety valves with subsequent inspection finding galled surfaces and damage to internal parts. Section 4.6 documented testing of Copes-Vulcan relief valves similar to the pressurizer PORVs at Byron and Braidwood, although the extent of water testing was not fully described. The report indicated no damage found during the inspection of the Copes-Vulcan relief valves. The report did not indicate any failures of the Crosby or Copes-Vulcan valves to reseat after discharging water during the testing.

EPRI also published NP-2770-LD in the early 1980s to describe the testing of PWR primary system safety valves. Volume 1, issued in December 1982, provides a summary of the test program and its results.[83] Section 4.5 of Volume 1 indicates that the following tests were performed on the Crosby 6M6 PSV: 11 steam tests with filled loop seals, 3 steam-to-water transition tests, and 2 water tests. The report states that the valve experienced chatter during the tests, and one water test had to be terminated. The individual volumes of EPRI NP-2770-LD discuss the test results for each specific PSV type. Volume 6, issued in March 1983, provides the test details for the Crosby 6M6 PSV.

Westinghouse Evaluation of EPRI Testing

In July 1982, the Westinghouse Owners Group (WOG) submitted WCAP-10105.[84] In WCAP-10105, the WOG indicated that the design specification for PSVs in Westinghouse-designed nuclear power plants is for steam service only. Based on a review of the EPRI test data, the WOG concluded that the valves performed with chatter, but did not identify any valve damage.

In January 1988, Westinghouse issued WCAP-11677, which compared the EPRI test data with feedwater line break safety analyses.[85] Westinghouse determined that all nuclear power plants addressed in the EPRI testing had PSVs that would operate reliably during water discharge. Westinghouse evaluated the performance of the Crosby 6M6 PSVs during the EPRI tests, and

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[80] EPRI 1982a
[81] VEPCO 2015
[82] EPRI 1982b
[83] EPRI 1982c
[84] WOG 1982
[85] Westinghouse 1988

considered that the performance involved less significant flutter (half lift motion) than the chatter (full lift motion) determined in the EPRI report. Westinghouse concluded that the Crosby 6M6 PSV can pass slightly subcooled water at a minimum up to three times without damage.

Byron and Braidwood Licensing and Response to TMI Requirements

The NRC safety evaluation reports (SERs) associated with the issuance of the operating licenses for Byron and Braidwood included evaluation of the TMI Action Plan items.[86] In the introduction to the Braidwood SER, the NRC staff stated that the review and evaluation of compliance by the applicant with the licensing requirements established in NUREG-0660[87] and TMI Action Plan Item II.D.1 were incorporated into the reviews summarized throughout the SER.

Appendix E, "Requirements Resulting from TMI-2 Accident," to the Byron and Braidwood UFSAR in Section E.23, "Relief and Safety Valve Test Requirements (II.D.1)," references the 1982 transmittal from Consumers Power of a test report for the EPRI safety and relief valve test program.[88] The UFSAR states that the final evaluation of the data indicated that the relief and safety valves will perform their intended functions for all expected fluid inlet conditions. The UFSAR also references the October 1982 licensee evaluation of the adequacy of the relief and safety valves that had been submitted to the NRC.[89]

In Supplement 1 to the Braidwood SER,[90] in Section 3.9.3.3, "Design and Installation of Pressure Relief Devices," the NRC staff stated that EPRI had completed a full-scale valve testing program and referenced the July 1982 submittal of WCAP-10105. The NRC staff stated that the applicant responded to a requirement to demonstrate operability of these valves through submittals dated July 1, 1982, October 26, 1982, and December 30, 1983. On the basis of a preliminary review, the NRC staff concluded that the applicant's general approach to responding to this item was acceptable, and provided adequate assurance that the RCS overpressure protection systems at Braidwood could adequately perform their intended functions. The NRC staff stated that if the detailed review revealed that modifications or adjustments to safety valves, PORVs, PORV block valves, or associated piping, would be needed to ensure that all intended design margins were present, the NRC staff would require that the applicant make appropriate modifications. The NRC staff categorized this issue as a Confirmatory Item. The NRC issued operating licenses for all four Byron and Braidwood Units between February 1985 and May 1988.

Closure of TMI Action Plan Item II.D.1 for Byron and Braidwood

Following the issuance of the operating licenses, the NRC staff documented its review of the response to TMI Action Plan Item II.D.1 for Byron and Braidwood via two letters that transmitted similar Technical Evaluation Reports (TERs) developed by Idaho National Engineering Laboratory (INEL).[91] In its letters, the NRC staff indicated that the licensee should develop and adopt plant procedures to inspect the pressurizer valves after each lift involving loop seal or water discharge. The TERs described the INEL review of the EPRI testing of PSVs and PORVs

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[86] NRC 1983 and NRC 1986b (Braidwood), NRC 1984 and NRC 1987a (Byron)
[87] NRC 1980a
[88] Consumers 1982
[89] ComEd 1982
[90] NRC 1986b. Similar discussion appears in NRC 1984 for Byron, and NRC 1987a (also for Byron) states that TMI Action Plan Item II.D.1 had been closed in NRC 1984.
[91] NRC 1988c (Byron) and NRC 1990a (Braidwood)

similar to the Byron and Braidwood pressurizer valves. The TERs concluded that Byron and Braidwood had provided an acceptable response to TMI Action Plan Item II.D.1.

Section 4.2.3, "Extended High Pressure Injection [HPI] Event," of the TERs stated that the potential for water discharge in extended HPI events can be disregarded for an extended high pressure injection event because at least 20 minutes would be available for operator action.

Water discharge was evaluated, however, in Section 4.2.2, "FSAR Liquid Transients," of the TERs. This section discussed the evaluation of the PSVs and PORVs for feedwater line break accidents that would include water discharge, and determined that the EPRI tests were applicable to the Byron and Braidwood PSVs and PORVs.

In addition, Section 4.3.1, "Safety Valves," and Section 4.3.2, "Power Operated Relief Valves," of the TERs determined that the performance of the PSVs and PORVs was acceptable based on the EPRI tests, including water discharge tests. The TERs indicated that the PSV had two applicable tests: a loop seal steam-water transition test where the valve opened, chattered and stabilized to close; and a saturated water test where the valve opened with water, chattered, and stabilized. The TERs indicated that the PORV opened and closed on demand in the loop seal steam-water transition test, with a bending moment that was evaluated by analysis.

APPENDIX C: CONCERNS REGARDING PERFORMANCE OF PRESSURIZER VALVES UNDER WATER FLOW CONDITIONS

Westinghouse Nuclear Safety Advisory Letter

In 1993 and 1994, Westinghouse issued Nuclear Safety Advisory Letter (NSAL) 93-013 its Supplement 1 to operating nuclear power plants (including Byron and Braidwood).[92] These advisories resulted from Westinghouse's discovery that potentially nonconservative assumptions were used in the licensing analysis of the Inadvertent Operation of the Emergency Core Cooling System at Power (IOECCS) event.

In NSAL-93-013, Westinghouse recommended that licensees determine whether their pressurizer safety relief valves (PSRVs) are capable of closing following discharge of subcooled water. Westinghouse noted that the PSRVs might have been designed or "qualified" to relieve subcooled water. Westinghouse indicated that water discharge through the power-operated relief valves (PORVs) is not a concern, because the PORV block valves can be used to isolate the PORVs if they fail to close. If the PSRVs are not designed or qualified for subcooled water discharge, Westinghouse recommended that licensees re-evaluate the IOECCS event with three possible options of (1) reducing emergency core cooling system (ECCS) flow used in the safety analysis, (2) using a less restrictive operator response time, or (3) crediting the use of one or more PORVs to help mitigate the event.

In Supplement 1 to NSAL-93-013, Westinghouse informed licensees of a potential reduced time for operator action if a positive displacement pump is in service. Westinghouse also advised licensees to qualify the PSRVs and the piping downstream of the PSRVs and PORVs if water discharge from the pressurizer were predicted.

Some licensees of operating nuclear power plants informed the NRC of their actions to address the potential concerns regarding water discharge from pressurizer safety valves (PSVs) and PORVs. A sample of actions by nuclear power plant licensees is summarized below in the "Plant-Specific Actions" section.

Additional NRC Generic Communications and Guidance

In 2003, the NRC staff issued a review standard for extended power uprate (EPU) reviews.[93] Item 8 on page 7 of the review standard states that pressurizer level should not be allowed to reach a pressurizer water-solid condition.

In 2005, the NRC issued Regulatory Issue Summary (RIS) 2005-29 to notify nuclear power plant licensees of a concern identified during reviews of power uprate requests.[94] In RIS 2005-29, the NRC staff stated that typically Condition II scenarios[95] involve discharging water through relief or safety valves that are not qualified for water discharge. The NRC staff stated that these valves are then assumed to fail in the open position and create a small-break loss-of-coolant accident (LOCA). The NRC staff stated that it was concerned that some licensees may be crediting PORVs without qualification for water discharge and without establishing additional restrictions to ensure the availability of PORVs and block valves. The NRC staff stated that the

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[92] Westinghouse 1993 and Westinghouse 1994
[93] NRC 2003
[94] NRC 2005b
[95] As defined in American Nuclear Society (ANS) Standard 51.1/N18.2-1973 (ANS 1973).

advice in Westinghouse NSAL-93-013 to use the PORV block valves to isolate the PORVs is inconsistent with non-escalation position.

In draft Revision 1 to RIS 2005-29, the NRC staff addresses the specific ANS Condition II scenarios of chemical volume and control system (CVCS) malfunction, inadvertent opening of a PORV or PSV (IOPSRV), and the IOECCS event.[96] Regarding the CVCS malfunction, the NRC staff states that performing only a reactivity anomaly analysis or assuming that this malfunction is not as severe as the IOECCS event is not acceptable. Regarding the IOPSRV event, the NRC staff stated that inadvertent opening of PSV or PORV could continue as an ANS Condition III small break LOCA and fails to meet the non-escalation position. Regarding the IOECCS event, the NRC staff states that five of the alternative approaches in NSAL-93-013 fail to meet the non-escalation position. The NRC staff indicated that these unacceptable alternative approaches are:

1. closing the block valve,
2. assuming that the PORV is not operable,
3. addressing a stuck-open PORV or PSV as a separate ANS Condition II event,
4. determining that a stuck-open PORV or PSV is not as severe as a small break LOCA, or
5. determining that RCS loss through PORV is made up by ECCS flow.

Additional General PSV/PORV Information

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.[97] 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.

In 2011, the NRC summarized relief valve performance data in NUREG/CR-7037, based on a study by the Idaho National Laboratory.[98] With respect to pressurizer PORVs, the report found four separate water discharge events at four PWR plants. The report estimated 698 total demands on these PORVs during their water discharge events with no failures to close. The report also summarized test data for three valve types from the Equipment Performance and Information Exchange (EPIX) database maintained by the Institute of Nuclear Power Operations. The report indicates two failures of PORVs to reclose during 2070 demands, but does not specify water or steam service for the EPIX test information. With respect to PSVs, the report indicates two failures out of four total demands following plant scrams, but does not indicate water or steam service. Following a request by the Panel, NRC staff from the Office of Nuclear Regulatory Research provided Licensee Event Report information indicating that the two PSV failures involved incomplete reseating of the valves with leakage of 25 and 200 gallons per minute, respectively. The report summarized EPIX test data for PSVs as no failures to reclose during 1805 demands.

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[96] NRC 2015a
[97] EPRI 2004
[98] NRC 2011

Plant-Specific Actions

Diablo Canyon

In 1996, the licensee for Diablo Canyon Power Plant (Diablo Canyon) submitted a report of its evaluation under Title 10 of the *Code of Federal Regulations* (10 CFR), Section 50.59, "Changes, tests and experiments," of the potential for an IOECCS event.[99] The submittal included NSAL-93-013 and its Supplement 1 as enclosures. The licensee indicated that the PSVs had not been initially qualified for water discharge, but were subsequently qualified to discharge water for a brief period. The licensee indicated that WCAP-11677 (which evaluated the EPRI testing) was applicable and demonstrated that the PSVs were operable.

In 2004, the NRC issued a license amendment for Diablo Canyon that allowed credit for actuation of the PORVs in response to inadvertent safety injection (SI) actuation, to avoid challenges to the PSVs.[100] To support the NRC staff's review, the licensee submitted additional information related to the capability of the PORVs to function adequately under conditions predicted for design-basis transients and accidents.[101] In response to a question regarding the design adequacy of the PORVs if the pressurizer becomes water solid, the licensee stated that the PORV had no requirements for ASME BPV Code certification, but referenced a January 1986 NRC letter that had accepted the adequacy of the PORV and block valve design and confirmatory testing for a range of fluid conditions (full pressure steam, steam to water transition, and subcooled water fluid).[102]

Salem

In 1997, the NRC issued a license amendment revising the technical specification (TS) for Salem Nuclear Generating Station, Units 1 and 2 (Salem) to ensure that the automatic capability of the PORVs to relieve pressure would be maintained.[103] In response to NSAL-93-013, the licensee determined that an inadvertent SI actuation at power could cause the pressurizer to become water solid. The PSVs would lift and discharge water if the automatic operation of the PORVs were not made available for reactor coolant system (RCS) depressurization early in the transient. In that the Salem PSVs were not designed to relieve water, it was noted that water discharge could cause the PSVs to fail in the open position.

During the review, the NRC staff noted that the PORVs were not designed to "safety related" standards and, thus, could not be credited for automatic mitigation of an inadvertent SI actuation at power. In response, the licensee proposed an upgrade of the PORVs to eliminate the possibility that a single active failure of a PORV component could prevent the mitigation of an inadvertent SI actuation at power. As discussed in the NRC staff's safety evaluation (SE), the licensee implemented modifications to the PORV circuitry to qualify the upgraded circuitry as safety-related.

Regarding PORV performance, the licensee evaluated the PORV air accumulators and determined that they had sufficient capacity for the inadvertent SI event. The licensee also reported that endurance tests had been performed with five different trims (with different trim

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[99] PG&E 1996
[100] NRC 2004a
[101] PG&E 2003
[102] NRC 1986a
[103] NRC 1997

materials) on one PORV at Wyle Laboratories to demonstrate that (1) after 2000 consecutive operations, there were no packing leaks or packing gland adjustments required; (2) there was no diaphragm failure; and (3) the solenoid valve withstood 10,000 operations without any loss of function. Based on this information, the NRC staff concluded that the PORV performance was acceptable to mitigate an inadvertent SI event.

Millstone 3

In 1998, the NRC issued a license amendment for Millstone Nuclear Power Station, Unit 3 (Millstone 3) that revised the TS to ensure that the capability of the PORVs to relieve pressure would be maintained.[104] The revised TS Bases stated that the PORVs and their associated piping had been demonstrated to be "qualified" for water discharge. The PORVs would prevent water discharge from the PSVs, for which qualification for water discharge had not been demonstrated. The TS Bases also stated that the prime importance for the capability to close the block valve is to isolate a stuck-open PORV. In the SE, the NRC staff referenced a December 1997 Licensee Event Report that notified the NRC of the issue of potential failure of PSVs following water discharge.[105]

As part of this license amendment, the licensee upgraded the PORV circuitry, added additional PORV surveillance requirements, qualified the PORVs and associated piping for water discharge, and revised emergency procedures to allow plant operators additional time to terminate the event. With respect to the PORV circuitry, the NRC staff concluded that the PORV circuitry modifications qualified the PORV control circuitry as safety-related. With respect to PORV performance, the licensee reanalyzed the inadvertent SI event with the LOFTRAN computer code to determine the time available for operator action to make a PORV available and provide the mass and energy releases needed to qualify the PORVs and associated piping for water discharge. The licensee referenced EPRI testing that was said to generically resolve TMI Action Plan Items associated with PORVs and safety valve qualification for water and steam discharge, specifically the results from four tests of a Garrett PORV (such as used at Millstone 3) for water discharge.[106] The licensee determined that the PORVs and associated piping are qualified for 1 hour of water discharge for an IOECCS event. The licensee also stated that the PORV manufacturer performed numerous cycle tests to verify the performance of the valve design, and also verified that valve seat leakage was acceptable. The licensee stated that the PORV block valves had been evaluated for water discharge in accordance with the program established in response to Generic Letter (GL) 89-10.[107] The NRC staff found the licensee information regarding the qualification of the PORVs for water discharge during the inadvertent SI event to be acceptable.

Callaway

In 2000, the NRC issued a license amendment for Callaway Plant, Unit 1 (Callaway) that revised the TS to change the PSV lift setting range.[108] The changes also credited automatic actuation of at least one PORV during an IOECCS event to prevent water discharge through the PSVs; to enable this credit, the licensee modified and upgraded the PORV circuitry to full Class 1E. In its license amendment request,[109] the licensee had stated that the design function

[104] NRC 1998
[105] Northeast 1997
[106] EPRI 1982a (Volume 11)
[107] NRC 1989
[108] NRC 2000

of the valves was not being changed and the conclusions documented in the NRC staff's previous evaluation of Callaway's response to TMI Action Plan Item II.D.1[110] were also unchanged. As a result, the licensee stated that the PORVs and associated discharge piping can accommodate water discharge.

Byron and Braidwood

In 1998, the licensee for Byron and Braidwood requested an amendment to its TS to take credit for automatic operation of the PORVs to mitigate an IOECCS event.[111] In the amendment request, the licensee stated that the PSVs had not been qualified to reseat after passing subcooled liquid. The licensee stated that the PORVs at Byron and Braidwood are safety-related components with safety-related actuators and accumulator tanks, with PORV control circuits classified as safety-related. The licensee noted that some portions of the PORV circuitry are nonsafety-related, with improvements implemented in response to GL 90-06.[112] The licensee stated that the PORV block valves are within the scope of the GL 89-10 program.

In 1999, the NRC staff requested additional information related to concerns that the PORV circuitry did not meet the single failure criterion.[113] The licensee reevaluated its approach and withdrew its TS amendment request.[114] No further action regarding this amendment request was identified by the Panel. However, in a public meeting during the review of the NRR Appeal,[115] the licensee stated that the PORVs and their block valves at Byron and Braidwood are safety-related with the exception of one circuitry aspect of the PORV.[116]

In 2001, the NRC issued a license amendment for Byron and Braidwood to increase the maximum thermal power for each unit from 3411 megawatts thermal (MWt) to 3586.6 MWt (commonly referred as a stretch power uprate).[117] During its review, the NRC staff requested that the licensee address water solid conditions in the pressurizer, because it had generally not accepted a solid pressurizer for an IOECCS event ~~to order~~ given the potential for all three PSVs to be stuck open due to liquid relief through these safety valves. In response, the licensee stated that Section 15.5.1, "Inadvertent Operation of Emergency Core Cooling System During Power Operation," of the UFSAR had been revised to credit the PSVs to pass water.[118] The licensee discussed the EPRI testing program in response TMI Action Plan Item II.D.1, with the results summarized in EPRI NP-2628-SR.[119] The licensee referenced previous NRC approvals related to TMI Action Plan Item II.D.1.[120]

The NRC staff made a further request regarding the temperature of water that would be discharged by the PSVs and the length of time that the PSVs would be expected to discharge water. The NRC staff also asked the licensee to discuss which EPRI tests are applicable to the Byron and Braidwood condition. In response, the licensee stated that the PSVs would close

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[109] Union Electric 2000
[110] NRC 1987b
[111] ComEd 1998
[112] NRC 1990b
[113] NRC 1999
[114] ComEd 1999
[115] Exelon 2015
[116] NRC 2016a
[117] NRC 2001b
[118] ComEd 2000b
[119] EPRI 1982b
[120] NRC 1998c and NRC 1990a

after discharging water, although they may not be leaktight.[121] The licensee stated that the leakage from up to three leaking PSVs is bounded by one fully open PSV. The licensee indicated that the EPRI testing of the Crosby safety valves in EPRI NP-2770-LD, Volumes 1 and 6,[122] are applicable. The licensee indicated that valve chatter occurred during the tests with damage to the internals, but that the safety valve closed in response to system depressurization. The licensee stated that the Byron and Braidwood pressurizer water temperature of 590 °F is higher than the EPRI tests (530 °F). The licensee stated that the assumed length of the event is 20 minutes from initial SI signal to when the system pressure is restored below PSV lift setpoint.

In Section 3.2 of the SE accompanying the license amendment, the NRC staff discussed its review of the performance of the PORVs and PSVs to discharge liquid water for approximately 20 minutes. The NRC staff discussed the EPRI testing program, with the conclusion that the PSV would close in response to system depressurization. The NRC staff reviewed the licensee's evaluation of the performance of the PSVs for liquid water conditions. The NRC staff found that the EPRI tests adequately demonstrated the performance of the valves for the expected water temperature conditions, and that there was reasonable assurance that the valves would adequately reseat following the spurious SI event. The NRC staff determined that EPRI test data indicated that the PSVs might chatter for the expected fluid inlet temperature, but that the resulting PSV seat leakage following the water discharge would be less than the discharge from one stuck-open PSV. Therefore, the NRC staff found the licensee's crediting of the PSVs to discharge liquid water during the spurious SI event to be acceptable. This portion of the SE was based on input provided by the Office of Nuclear Reactor Regulation (NRR) Reactor Systems Branch, with technical input from the responsible staff member for safety valves in the NRR Division of Engineering.[123]

As noted by the licensee, Section 15.5.1 of the Byron and Braidwood UFSAR at the time of the stretch power uprate includes PSV water discharge and references the TMI Action Plan Item II.D.1 approvals.[124] The current UFSAR Revision 15 concludes that the IOECCS event does not progress into a stuck-open PSV LOCA event.[125] The UFSAR states that all three PSVs may lift but will reclose, and that the leakage is bounded by one fully open valve with the consequences bounded by the IOPSRV event. The UFSAR also specifies that if SI results in discharge of coolant through the pressurizer valves, the operators will bring the plant to cold shutdown to inspect the valves.

In 2004, the NRC issued a license amendment for Byron and Braidwood granting an adjustment to the PSV setpoints.[126] As documented in the SE, the NRC staff requested during its review that the licensee perform a quantitative analysis regarding PSV water cycles and discharge water temperature. For the loss of ac power (LOAC) with reactor coolant pump (RCP) seal injection event, the licensee's analysis indicated that continued injection of water into the RCS through the RCP seals would result in a water-solid pressurizer and water discharge through the PSVs. The proposed PSV setpoint tolerance assuming negative tolerance would result in a lower PSV lift setpoint. With the lower setpoint, the PSV would open earlier, and a larger number of PSV water cycles with a lower water discharge temperature could result during the

[121] Exelon 2001
[122] EPRI 1982c and EPRI 1983
[123] NRC 2001a
[124] Exelon 2002
[125] Exelon 2014
[126] NRC 2004b

transient. The licensee performed an analysis of the LOAC with RCP seal injection event, and determined the revised PSV setpoint would result in an increase of about one PSV water cycle and a reduction in the water discharge temperature of about 0.5 °F. A comparison of the reanalysis showed that the spurious SI event remained the limiting event since it resulted in a greater increase in the number of PSV water cycles (two cycles vs. one cycle) and a greater decrease in the PSV discharge water temperature (3.0 °F vs. 0.5 °F) than that calculated for the LOAC with RCP seal injection event. The water discharge temperature in the analysis of record for the spurious SI event was 590 °F. The lowest discharge water temperature for the spurious SI event with the revised PSV setpoint was 587 °F. The NRC staff found that the calculated water discharge temperature (587 °F) was significantly higher than the discharge water temperature of 530 °F that was used to support operability of the PSVs as discussed in the analysis of record. As a result, the NRC staff concluded that the analysis was acceptable to assure that the PSVs will remain operable following a spurious SI event.

In 2014, the NRC issued a license amendment for Byron and Braidwood granting a measurement uncertainty recapture (MUR) power uprate.[127] The NRC staff determined that the IOECCS event was outside of the scope of the MUR power uprate, because the licensee did not propose to modify the Chapter 15 analyses related to PSV and PORV water discharge.

With respect to inservice testing (IST) activities, the Byron IST program[128] references the ASME *Code for Operation and Maintenance of Nuclear Power Plants* (OM Code), 2004 Edition through 2006 Addenda; and the Braidwood IST program[129] references the ASME OM Code, 2001 Edition through 2003 Addenda. The Byron IST Program specifies the following testing and intervals for the PORVs, PORV block valves, and PSVs:

- **PORV:** fail safe test closed (cold shutdown interval), stroke-time exercise open and closed (cold shutdown interval), and position indication test (2 year interval)

- **PORV Block Valve:** exercise open and closed (2 year interval); position indication test (Joint Owners Group (JOG) Program interval); and open and closed test in accordance with ASME OM Code Case OMN-1, "Alternative Rules for Preservice and Inservice Testing of Active Electric Motor Operated Valve Assemblies in Light-Water Reactor Power Plants" (JOG Program interval)

- **PSV:** position indication test (2 year interval) and relief valve test (5 year interval), referencing ASME OM Code, Appendix I, "Inservice Testing of Pressure Relief Devices in Light-Water Reactor Nuclear Power Plants"

The Braidwood IST Program specifies the following testing and intervals for the PORVs, PORV block valves, and PSVs:

- **PORV:** fail safe test closed (refueling outage interval), stroke-time exercise open and closed (refueling outage interval), and position indication test (2 year interval).

- **PORV Block Valve:** exercise open and closed (quarterly interval) and position indication test (2 year interval)

[127] NRC 2014a
[128] Exelon 2016b
[129] Exelon 2009

- **PSV:** position indication test (2 year interval), and relief valve test (5 year interval), referencing ASME OM Code, Appendix I

Shearon Harris

In 2001, the NRC issued a license amendment to Shearon Harris Nuclear Power Plant, Unit 1 (Shearon Harris) for steam generator replacement and a power uprate to a maximum power level of 2900 MWt (approximately 4.5 percent).[130] In addressing the licensee's evaluation of Standard Review Plan (SRP) Section 15.5.1, the NRC staff found that the analysis showed that the calculated inlet pressures and temperatures required for the PORVs and safety relief valves (SRVs)[131] to operate in a water environment were within the valve operable ranges, and thus ensured that the PORV and SRV would be operable during the transient. The valve operable ranges were previously determined by the licensee to support operability of the PORV and SRV during the discharge of subcooled water in accordance with the TMI Action Plan Item II.D.1 requirements. Based on the analysis meeting the acceptance criteria of SRP Section 15.5.1 with respect to the RCS pressure limit and departure-from-nucleate-boiling limit, the NRC staff concluded that the analysis was acceptable.

Beaver Valley

In 2006, the NRC issued a license amendment authorizing an EPU for Beaver Valley Power Station, Units 1 and 2 (Beaver Valley), an approximate 8-percent increase in thermal power to 2,900 MWt.[132] In the SE accompanying the amendment, the NRC staff described its review of the capability of the PSVs to discharge liquid and adequately reseat for a spurious SI actuation. The NRC staff specifically evaluated whether the PSVs could reasonably be expected to reseat to prevent the spurious SI actuation (an ANS Condition II event) from causing a stuck-open PSV (an ANS Condition III event). This issue was said to be further discussed in RIS 2005-29. While the PSVs for Beaver Valley were qualified to discharge steam, if the valves discharged water with sufficient subcooling, the NRC staff was concerned that they might not reseat properly.

Based the licensee's analysis, during a spurious SI event, the PSVs would be required to discharge steam followed by high temperature water after the pressurizer filled. The licensee provided plots of the pressurizer water temperatures for this event that indicated that the minimum temperature of the discharged liquid for Beaver Valley was approximately 620 °F. To evaluate the capability of the valves to discharge and reseat, the NRC staff reviewed the available data from the full-flow tests performed during the EPRI test program in 1981 for the specific PSV models representative of those installed at Beaver Valley. The licensee also used the methodology contained in WCAP-11677 and determined that the minimum acceptable liquid temperature for which the PSVs were expected to successfully discharge and reseat was less than the minimum expected temperature for the spurious SI event for Beaver Valley.

The NRC staff agreed that both the minimum expected water discharge temperature and the minimum acceptable water temperature had been conservatively calculated. Therefore, the NRC staff determined that, for purposes of preventing the occurrence of a more serious ANS Condition III event, there was reasonable assurance that the PSVs would discharge water and

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130 NRC 2001d

131 This term is used in the Shearon Harris SE~~; the Panel considers the term SRV to be equivalent to PSV for this facility~~. The licensee's RAI response (CP&L 2000) makes clear that the referenced SRVs and PORVs are pressurizer valves.

132 NRC 2006

reseat adequately following a spurious SI actuation. A consideration of the NRC staff in making this finding was that, in the unlikely event of a stuck-open PSV, the ECCS was fully capable of mitigating the resulting LOCA.

Turkey Point

In 2012, the NRC issued a license amendment authorizing an EPU for Turkey Point Nuclear Generating, Units 3 and 4 (Turkey Point), increasing the thermal power level of each unit approximately 15 percent to 2644 MWt.[133]

In the SE accompanying the amendment, the NRC staff indicated that ECCS actuation was not a possible initiator of inadvertent increase in reactor coolant inventory because the high head SI pumps have a shut-off head below the normal RCS operating pressure. The NRC staff stated that a CVCS malfunction that increases RCS inventory was evaluated for the effects of adding water inventory to the RCS. If the pressurizer filled and caused water to be relieved through the PORVs or PSVs, then these valves could stick open and create a small break LOCA. The NRC staff stated that this would violate the acceptance criterion that prohibits the escalation of an anticipated operational occurrence (AOO) into a more serious event. Satisfaction of this acceptance criterion was demonstrated by showing that sufficient time would exist for the operator to recognize the situation and end the charging flow before the pressurizer could fill. The NRC staff concluded that the licensee's analyses of IOECCS and CVCS events adequately accounted for operation of the plant at the proposed power level.

Regarding an inadvertent opening of a PORV, the licensee initially proposed that the consequences of this event were bounded by the small break LOCA. The NRC staff did not accept this proposed disposition. If action were not taken to secure the open valve by either closing the PORV or its block valve, the NRC staff stated that this event could escalate to a small-break LOCA, which would be contrary to the non-escalation position. When the pressurizer filled, water would begin to flow through the open PORV. If the PORV were not qualified for water discharge, the NRC staff stated that it was likely the PORV would not close upon demand. In this way, the NRC staff stated that the inadvertent opening of a PORV, an AOO, would become a small break-LOCA at the top of the pressurizer, an ANS Condition III event. The NRC staff requested that the licensee address the inadvertent opening of the PORV with respect to the third criterion for an ANS Condition II event.

The licensee provided an analysis performed largely in accordance with NRC-approved, Westinghouse analytic methodology using the RETRAN computer code; however, this analysis was performed assuming that the PORV opened instead of the PSV. The NRC staff stated that assuming the opening of the PORV is acceptable, because the PSV is differently qualified, and reseats mechanically. An additional independent fault would be required to cause the PSV to fail to close. The analysis indicated that the pressurizer would fill within about 240 seconds. The licensee stated that there were multiple alarms to indicate the opening of a PORV. The licensee stated that a prompt operator action would be needed to close the PORV and, if the PORV does not close, the operator would be directed to close the block valve. Because the necessary actions would be prompt and simple, the NRC staff agreed that there would be sufficient time to secure the inadvertently open PORV without filling the pressurizer.

[133] NRC 2012a

St. Lucie

In 2012, the NRC issued a license amendment authorizing an EPU for St. Lucie Plant, Unit 2 (St. Lucie, Unit 2) that increased the authorized thermal power level about 12 percent to 3020 MWt.

Regarding an IOECCS event, the high pressure SI pumps would be incapable during power operations of delivering flow to the RCS because the pumps' shut-off head would be less than the normal RCS operating pressure of 2250 pounds per square inch absolute. Therefore, the licensee determined that the inadvertent operation of the ECCS at power event was not a credible event and did not analyze it for the proposed EPU. The NRC staff found that the licensee's position for not analyzing the IOECCS event to be acceptable.

Regarding a CVCS malfunction, the licensee evaluated it as an AOO for the effects of adding water inventory to the RCS. The NRC staff reviewed the licensee's analyses of the CVCS malfunction event and concluded that the licensee's analyses adequately accounted for operation of the plant at the proposed power level and were performed using acceptable analytical models. The NRC staff determined that the licensee's analysis demonstrated that the pressurizer did not become water solid, assuring no water was discharged through the PSVs.

Regarding an IOPORV event, the NRC staff stated that, when viewed from the mass addition perspective, this event could be evaluated in two phases: (1) an inadvertent opening of a pressurizer relief valve, followed by (2) an inadvertent ECCS actuation. In the first phase, the NRC staff stated that this event could be mitigated by closing the open PORV or its block valve. If the PORV or its block valve was not closed, the NRC staff stated that the IOPORV event would enter the second phase with actuation of the ECCS. Based on its review, the NRC staff determined that the pressurizer overfill analysis, available alarming system, and procedures, in combination with simulator exercise results, provided reasonable assurance that the pressurizer would not be expected to fill to a water solid condition that could prevent the PORV or PSV from closing after they were open. The NRC staff therefore concluded that the event would not generate a more serious plant condition, meeting the non-escalation criterion. The NRC staff stated that it reviewed the licensee's analyses of the inadvertent opening of a pressurizer PORV event, and concluded that the licensee's analyses adequately accounted for operation of the plant at the proposed power level and were performed using acceptable analytical models.

The NRC staff concluded that the licensee demonstrated that all AOO acceptance criteria were satisfactorily met.

North Anna

North Anna Power Station, Units 1 and 2 (North Anna) UFSAR Section 15.2.14, "Spurious Operation of the Safety Injection System at Power," describes plant response to an inadvertent SI event. In particular, UFSAR Section 15.2.14.2.3, "Event Propagation," states the following:

> Safety valve (Reference 18) and PORV (Reference 19) testing has revealed no instances of failure of the valves to reseat following water relief. Resulting leakage is within the capacity of the normal makeup system and is therefore not considered to be a small break loss of reactor coolant event. Therefore, the complete filling of the pressurizer and/or water relief via a safety valve as a result of a spurious safety injection does not constitute a failure to meet the event propagation acceptance criterion. Although primary credit for preventing the propagation of the event to a small break loss of reactor coolant event is the

reseating of the PORVs and safety valves, it is noted that the PORVs (which open prior to the safety valves and, if open, preclude safety valve actuation for this event) are provided with block valves which the operator will close in the event of excessive PORV leakage.

North Anna UFSAR Section 15.2.14.3, "Conclusions," indicates that the complete filling of the pressurizer and water discharge via a PSV as a result of a spurious SI does not constitute a failure to meet the non-escalation position. Furthermore, UFSAR Section 15.2, "References," lists EPRI NP-2770-LD and EPRI NP-2670-LD.

Conclusion

In conclusion, the reliance by the licensee for Byron and Braidwood on the acceptable performance of the PSVs and PORVs following water discharge in response to abnormal events is not inconsistent with similar approaches by some other nuclear power plant licensees. In general, the review of activities by various nuclear power plant licensees related to PSV and PORV performance revealed reliance on EPRI, Wyle, and valve vendor testing to provide support for the performance of these valves under various service conditions. Specific certification for flow capacity of these valves for water discharge in accordance with the ASME BPV Code and National Board was not identified in the review of various justifications prepared by nuclear power plant licensees.

In evaluating the historical documents for Byron and Braidwood, the Panel found it challenging to determine specifically how the licensee resolved the concern raised in NSAL-93-013 in its analyses and plant operations. While the record does not currently support a compliance backfit in this case, if (as recommended by the Panel) the NRC staff undertakes a generic review of licensees' treatment of the potential for pressurizer valve damage following water discharge, it may be appropriate to consider what actions have been taken, how operating experience with water discharge has been considered, and how analysis assumptions are considered in operational practices (including inservice testing) at each plant.

APPENDIX D: REFERENCES

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4. **ANS 1981:** ANS and American National Standards Institute (ANSI), ANSI/ANS 58.9, "Single Failure Criteria for Light Water Reactor Safety-Related Fluid Systems," issued February 17, 1981 (updated in 2002 and reaffirmed in 2015). [Publicly available from ANS for a fee.]

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11. **Consumers 1982:** Consumers Energy Company, letter from D. P. Hoffman to Harold R. Denton, U.S. NRC, "Safety and Relief Valve Test Report for the EPRI PWR Safety and Relief Valve Test Program," dated April 1, 1982. [This document is not in ADAMS, but is available through the NRC Public Document Room using Accession No. 8207160337 and microfiche location 13866:001 – 13869:106.]

12. **CP&L 2000:** Carolina Power & Light Company, letter from James Scarola to U.S. NRC, "License Amendment Application Power Uprate," dated December 14, 2000. ADAMS Accession No. ML003780761.

13. **EPRI 1982a:** EPRI, NP-2670-LD, "EPRI/Wyle Power-Operated Relief Valve Phase III Test Report," dated October 1982. [Publicly available from EPRI for a fee.]

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19. **Exelon 2002:** Exelon Generation Company, LLC, letter from R. M. Krich to U.S. NRC, "Byron/Braidwood Stations Updated Final Safety Analysis Report, Revision 9; Byron Station Technical Requirements Manual, Revision 32; Byron Station Technical Specifications Bases, Revision 31; Braidwood Station Technical Requirements Manual, Revision 26; Braidwood Station Technical Specifications Bases, Revision 36," dated December 16, 2002. [Non-publicly available; electronic files are in the NRC file center, and the transmittal letter is at ADAMS Accession No. ML023650683.]

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33. **NRC 1980b:** U.S. NRC, letter from Darrell G. Eisenhut, "Post-TMI Requirements," dated October 31, 1980. Text available through ADAMS Public Legacy Library using Accession No. 8012160050.

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38. **NRC 1982:** U.S. NRC, memorandum from R.L. Tedesco to R.F. Fraley, "Issuance of the Draft Safety Evaluation Report for the Watts Bar Nuclear Plant, Units 1 and 2," dated January 5, 1982. ADAMS Accession No. ML073580524 [non-public].

39. **NRC 1983:** U.S. NRC, NUREG-1002, "Safety Evaluation Report Related to the Operation of Braidwood Station, Units 1 and 2," dated November 1983. ADAMS Accession No. ML12114A272.

40. **NRC 1984:** U.S. NRC, NUREG-0876, Supplement No. 5, "Safety Evaluation Report Related to the Operation of Byron Station, Units 1 and 2," dated October 1984. ADAMS Accession No. ML091340252.

41. **NRC 1985:** U.S. NRC, "Revision of Backfitting Process for Power Reactors - Parts 2 and 50," dated September 20, 1985. 50 FR 38097.

42. **NRC 1986a:** U.S. NRC, letter from H. Schierling to J.D. Shiffer, Pacific Gas and Electric Company (PG&E), "Safety and Relief Valve Testing, NUREG-0737 Item II.D.1," dated January 27, 1986. [This document is not in ADAMS, but is available through the NRC Public Document Room using Accession No. 8602180450 and microfiche location 34630:271-299.]

43. **NRC 1986b:** U.S. NRC, NUREG-1002, Supplement No. 1, "Safety Evaluation Report Related to the Operation of Braidwood Station, Units 1 and 2," dated September 1986. [This document is not in ADAMS, but is available through the NRC Public Document Room using Accession No. 8610220157 and microfiche location 38332:281 – 38333:087.]

44. **NRC 1987a:** U.S. NRC, NUREG-0876, Supplement No. 8, "Safety Evaluation Report Related to the Operation of Byron Station, Units 1 and 2," dated March 1987. [This document is not in ADAMS, but is available through the NRC Public Document Room using Accession No. 8704010167 and microfiche location 40321:134-156.]

45. **NRC 1987b:** U.S. NRC, letter from Thomas W. Alexion to D.F. Schnell, Union Electric Company, dated September 10, 1987. [This document is not in ADAMS, but is available through the NRC Public Document Room using Accession No. 8709170429 and microfiche location 42712:212-229.]

46. **NRC 1988a:** U.S. NRC, SECY-88-102, "Final Backfit Rule," dated April 18, 1988. [This document is not in ADAMS, but is available through the NRC Public Document Room using Accession No. 8805130269 and microfiche location 45795:299-362.]

47. **NRC 1988b:** U.S. NRC, "Revision of Backfitting Process for Power Reactors - Final Rule," dated June 6, 1988. 53 FR 20603.

48. **NRC 1988c:** U.S. NRC, letter from L. Olshan to Henry E. Bliss, Commonwealth Edison Company, "NUREG-0737, Item II.D.1, Performance Testing on Relief and Safety Valves for Byron Station, Units 1 and 2," dated August 18, 1988. ADAMS Accession No. ML003772409 [pages 161-188 of file].

49. **NRC 1988d:** U.S. NRC, NRC Manual Chapter 0514, "NRC Program for Management of Plant-Specific Backfitting of Nuclear Power Plants," dated August 26, 1988. ADAMS Accession No. ML041400111.

50. **NRC 1988e:** U.S. NRC, Bulletin 88-11, "Pressurizer Surge Line Thermal Stratification," dated December 20, 1988. Publicly available on the NRC website.

51. **NRC 1989:** U.S. NRC, Generic Letter (GL) 89-10, "Safety-Related Motor-Operated Valve Testing and Surveillance," dated June 28, 1989. ADAMS Accession No. ML031150300.

52. **NRC 1990a:** U.S. NRC, letter from S. Sands to Thomas J. Kovach, Commonwealth Edison Company, "NUREG-0737, Item II.D.1, "Performance Testing on Relief and Safety Valves for Braidwood Station, Units 1 and 2," dated May 21, 1990. ADAMS Accession No. ML003772409 [pages 189-217 of file].

53. **NRC 1990b:** U.S. NRC, GL 90-06, "Resolution of Generic Issue 70, 'Power-Operated Relief Valve and Block Valve Reliability' and Generic Issue 94, 'Additional Low-Temperature Over Pressure Protection for Light-Water Reactors,' Pursuant to 10 CFR 50.54(f)," dated June 25, 1990. ADAMS Accession No. ML031210416.

54. **NRC 1990c:** U.S. NRC, NUREG-1409, "Backfitting Guidelines," dated July 1990. ADAMS Accession No. ML032230247.

55. **NRC 1991:** U.S. NRC, letter from Anthony H. Hsia to Thomas J. Kovach, Commonwealth Edison Company, "Issuance of Amendments [for Byron and Braidwood] (TAC Nos. M77332, M77333, M77334, M77335, M77402, M77403, M77404, and M77405)," dated November 18, 1991. ADAMS Accession No. ML020860105.

56. **NRC 1994a:** U.S. NRC, SECY-94-084, "Policy and Technical Issues Associated with the Regulatory Treatment of Non-Safety Systems in Passive Plant Designs," dated March 28, 1994. ADAMS Accession No. ML003708068.

57. **NRC 1994b:** U.S. NRC, SRM-SECY-94-084, "SECY-94-084 – Policy and Technical Issues Associated with the Regulatory Treatment of Non-Safety Systems in Passive

Plant Designs and COMSECY-94-024 – Implementation of Design Certification and Light-Water Reactor Design Issues," dated June 30, 1994. ADAMS Accession No. ML003708098.

58. **NRC 1996:** U.S. NRC, NUREG-0800, SRP Section 15.5.1 – 15.5.2, "Inadvertent Operation of ECCS and Chemical and Volume Control System Malfunction that Increases Reactor Coolant Inventory," draft Revision 2, dated April 1996. ADAMS Accession No. ML052070725.

59. **NRC 1997:** U.S. NRC, letter from Leonard N. Olshan, NRC, to Leon R. Eliason, Public Service Electric & Gas Company, "Salem Nuclear Generating Station, Unit Nos. 1 and 2 (TAC Nos. M97827 and M97828)," dated June 4, 1997. ADAMS Accession No. ML011720397.

60. **NRC 1998:** U.S. NRC, letter from James W. Andersen to Martin L. Bowling, Jr., Northeast Nuclear Energy Company, "Issuance of Amendment – Millstone Nuclear Power Station, Unit No. 3 (TAC No. MA1527)," dated June 5, 1998. ADAMS Accession No. ML011800207.

61. **NRC 1999:** U.S. NRC, letter from John B. Hickman to Oliver D. Kingsley, Commonwealth Edison Company, "Request for Additional Information – Byron Station, Units 1 and 2 and Braidwood Station, Units 1 and 2 (TAC Nos. MA2043, MA2044, MA2048, and MA2049," dated May 13, 1999. [This document is not in ADAMS, but is available through the NRC Public Document Room using Accession No. 9905170241 and microfiche location A8035:313-316.]

62. **NRC 2000:** U.S. NRC, letter from Jack Donohew to Garry L. Randolph, Union Electric Company, "Callaway Plant, Unit 1 – Issuance of Amendment Re: Pressurizer Safety Valves and Power Operated Relief Valves (PORVs) (TAC No. MA9080)," dated September 26, 2000. ADAMS Accession No. ML003753326.

63. **NRC 2001a:** U.S. NRC, memorandum from Frank Akstulewicz to Anthony Mendiola, "Byron and Braidwood Stations, Units 1 and 2 – Requests for a License Amendment to Permit Uprated Power Operations (TAC Nos. MA9426, MA9427, MA9428 and MA9429)," dated March 15, 2001. ADAMS Accession No. ML010740316 [non-public].

64. **NRC 2001b:** 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.

65. **NRC 2001c:** U.S. NRC, NUREG-1800, "Standard Review Plan for Review of License Renewal Applications for Nuclear Power Plants," dated July 2001. ADAMS Accession No. ML012070413.

66. **NRC 2001d:** U.S. NRC, letter from N. Kalyanam to James Scarola, Carolina Power & Light Company, "Shearon Harris Nuclear Power Plant, Unit 1 – Issuance of Amendment Re: Steam Generator Replacement and Power Uprate (TAC Nos. MB0199 and MB0782)," dated October 12, 2001. ADAMS Accession No. ML012880381.

67. **NRC 2003:** U.S. NRC, Review Standard (RS) 001, "Review Standard for Extended Power Uprates," dated December 2003. ADAMS Accession No. ML033640024.

68. **NRC 2004a:** U.S. NRC, letter from Girija S. Shukla to Gregory M. Rueger, PG&E, "Diablo Canyon Power Plant, Unit Nos. 1 and 2 – Issuance of Amendment Re: Credit for Automatic Actuation of Pressurizer Power Operated Relief Valves (TAC Nos. MB6758 and MB6759)," dated July 2, 2004. ADAMS Accession No. ML041950300.

69. **NRC 2004b:** U.S. NRC, letter from George F. Dick, Jr., to Christopher M. Crane, Exelon Generation Company, LLC, "Byron Station, Units 1 and 2, and Braidwood Station, Units 1 and 2 – Issuance of Amendments, Re: Pressurizer Safety Valve Setpoints, (TAC Nos. MB9762, MB9763, MB9760, and MB9761)," dated August 26, 2004. ADAMS Accession No. ML042250531.

70. **NRC 2005a:** U.S. NRC, SECY-05-0138, "Risk-Informed and Performance-Based Alternatives to the Single-Failure Criterion," dated August 4, 2005. ADAMS Accession No. ML051950610.

71. **NRC 2005b:** U.S. NRC, Regulatory Issue Summary 2005-29, "Anticipated Transients that Could Develop into More Serious Events," dated December 14, 2005. ADAMS Accession No. ML051890212.

72. **NRC 2006:** U.S. NRC, letter from Timothy G. Colburn to James H. Lash, FirstEnergy Nuclear Operating Company, "Beaver Valley Power Station, Unit Nos. 1 and 2 (BVPS-1 and 2) – Issuance of Amendment Regarding the 8-Percent Extended Power Uprate (TAC Nos. MC4645 and MC4646)," dated July 19, 2006. ADAMS Accession No. ML061720274.

73. **NRC 2007a:** U.S. NRC, NUREG-0800, SRP Section 15.0, "Introduction - Transient and Accident Analyses," Revision 3, dated March 2007. ADAMS Accession No. ML070710376.

74. **NRC 2007b:** U.S. NRC, NUREG-0800, SRP Section 15.5.1 – 15.5.2, "Inadvertent Operation of ECCS and Chemical and Volume Control System Malfunction that Increases Reactor Coolant Inventory," Revision 2, dated March 2007. ADAMS Accession No. ML070820081.

75. **NRC 2007c:** U.S. NRC, NUREG-0800, SRP Section 15.6.1, "Inadvertent Opening of a PWR Pressurizer Pressure Relief Valve or a BWR Pressure Relief Valve," Revision 2, dated March 2007. ADAMS Accession No. ML070820094.

76. **NRC 2011:** U.S. NRC, NUREG/CR-7037, "Industry Performance of Relief Valves at U.S. Commercial Nuclear Power Plants through 2007," dated March 2011. ADAMS Accession No. ML110980205.

77. **NRC 2012a:** U.S. NRC, letter from Jason C. Paige to Mano Nazar, Florida Power and Light Company, "Turkey Point Units 3 and 4 - Issuance of Amendments Regarding Extended Power Uprate (TAC Nos. ME4907 and ME4908)," dated June 15, 2012. ADAMS Accession No. ML11293A359.

78. **NRC 2012b:** U.S. NRC, letter from Tracy J. Orf to Mano Nazar, Florida Power and Light Company, "St. Lucie Plant, Unit 2 - Issuance of Amendment Regarding Extended Power

Uprate (TAC No. ME5843)," dated September 24, 2012. ADAMS Accession No. ML12268A132.

79. **NRC 2013:** U.S. NRC, Management Directive 8.4, "Management of Facility-Specific Backfitting and Information Collection," dated October 9, 2013. ADAMS Accession No. ML12059A460.

80. **NRC 2014a:** 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.

81. **NRC 2014b:** U.S. NRC, memorandum from Samuel Miranda to Christopher P. Jackson, "Making Non-Concurrence NCP-2013-04 Public," dated February 28, 2014. ADAMS Accession No. ML14063A174.

82. **NRC 2015a:** U.S. NRC, Draft Revision 1 to RIS 2005-29, "Anticipated Transients that Could Develop into More Serious Events," dated July 13, 2015. ADAMS Accession No. ML15014A469. (Also published at for public comment at 80 FR 42559.)

83. **NRC 2015b:** U.S. NRC, letter from Anne T. Boland to Bryan Hanson, Exelon Generation Company, LLC, "Braidwood Station, Units 1 and 2, and Byron Station, Unit Nos. 1 and 2 – Backfit Imposition Regarding Compliance with 10 CFR § 50.34(b), GDC 15, GDC 21, GDC 29, and Licensing Basis (TAC Nos. MF3206, MF3207, MF3208, and MF3209)," dated October 9, 2015. ADAMS Accession No. ML14225A871.

84. **NRC 2016a:** U.S. NRC, "Official Transcript of Proceedings – Public Meeting to Discuss Exelon Generating Company, LLC's Appeal of Compliance Backfit Affecting Byron and Braidwood Generating Stations," dated March 7, 2016. ADAMS Accession No. ML16070A364.

85. **NRC 2016b:** U.S. NRC, memorandum from Anthony T. Gody, Jr., to Marissa G. Bailey, "Input for Exelon Backfit Review Panel," dated March 21, 2016. ADAMS Accession No. ML16081A405 [non-public].

86. **NRC 2016c:** U.S. NRC, memorandum from Marissa G. Bailey to William M. Dean, "Backfit Review Panel Recommendation Regarding Exelon Appeal of Backfit Affecting Byron and Braidwood Stations Regarding Compliance with 10 CFR 50.34(b), GDC 15, GDC 21, GDC 29, and the Licensing Basis," dated March 25, 2016. ADAMS Accession No. ML16082A542 [non-public].

87. **NRC 2016d:** U.S. NRC, letter from William M. Dean to J. Bradley Fewell, Exelon Generation Company, LLC, "U.S. Nuclear Regulatory Commission Response to Backfit Appeal - Braidwood Station, Units 1 and 2, and Byron Station, Units 1 and 2," dated May 3, 2016. ADAMS Accession No. ML16095A204.

88. **NRC 2016e:** U.S. NRC, memorandum from Victor M. McCree to Gary M. Holahan, K. Steven West, Thomas G. Scarbrough, and Michael A. Spencer, "Charter for Backfit Appeal Review Panel Associated with Byron and Braidwood Compliance with 10 CFR 50.34(b), GDC 15, GDC 21, GDC 29, and the Licensing Basis," dated June 22, 2016. ADAMS Accession No. ML16173A311 [non-public].

89. **NRC 2016f:** U.S. NRC, "An Assessment of Core Damage Frequency for Byron/Braidwood Nuclear Power Plants Supporting Backfit Appeal Review Panel," dated August 11, 2016. ADAMS Accession No. ML16214A199 [non-public].

90. **PG&E 1996:** PG&E, letter from Gregory M. Rueger to U.S. NRC, "Forwards completed Licensing Basis Impact Evaluation of FSAR Update change which contains SE performed IAW 10CFR50.59 re reanalysis of inadvertent ECCS actuation accident," dated August 13, 1996. [This document is not in ADAMS, but is available through the NRC Public Document Room using Accession No. 9608200112 and microfiche location 89419:294-322.]

91. **PG&E 2003:** Pacific Gas and Electric Company, letter from David H. Oatley to U.S. NRC, "Response to NRC Request for Additional Information Regarding License Amendment Request 01-018, 'Credit for Automatic Actuation of Pressurizer Power Operated Relief Valves; Pressurizer Safety Valve Loop Seal Temperature,'" dated November 21, 2003. ADAMS Accession No. ML033360735.

92. **TVA 1983:** Tennessee Valley Authority (TVA), letter from L.M. Mills to E. Adensam, U.S. NRC, enclosing "Watts Bar Nuclear Plant Units 1 and 2 Safety Valve Sizing," dated April 18, 1983. ADAMS Accession No. ML080360226.

93. **Union Electric 2000:** Union Electric Company, letter from Alan C. Passwater to U.S. NRC, "Revision to Technical Specifications 3.3.2, 3.4.10, and 3.4.11 – Pressurizer Safety Valves and PORVs," dated May 25, 2000. ADAMS Accession No. ML003719636.

94. **VEPCO 2009:** Virginia Electric and Power Company (VEPCO), letter from L.N. Hartz to U.S. NRC, "North Anna Power Station Units 1 and 2 Updated Final Safety Analysis Report Revision 45," dated October 1, 2009. ADAMS Accession No. ML092810047.

95. **VEPCO 2015:** Virginia Electric and Power Company, letter from Gianna C. Clark to U.S. NRC, "North Anna Power Station Units 1 and 2 Updated Final Safety Analysis Report Revision 51," dated September 30, 2015. ADAMS Accession No. ML15296A098 [non-public].

96. **Westinghouse 1988:** Westinghouse, R.J. Dickinson and J.G. Bass, "Pressurizer Safety Relief Valve Operation for Water Discharge during a Feedwater Line Break," WCAP-11677, dated January 1988. (Submitted to the NRC as part of a May 8, 1989, response to a request for additional information related to Seabrook.) [This document is not in ADAMS, but is available through the NRC Public Document Room using Accession No. 8905120191 and microfiche location 49755:336 – 49756:017.]

97. **Westinghouse 1993:** Westinghouse, Nuclear Safety Advisory Letter (NSAL) 93-013, "Inadvertent ECCS Actuation at Power," dated June 30, 1993. [This document is not in public ADAMS, but is available through the NRC Public Document Room using Accession No. 9608200112 and microfiche location 89419:311-315, as well as in non-public ADAMS Accession No. ML052930330, pages 2-6 of file.]

98. **Westinghouse 1994:** Westinghouse, NSAL-93-013, Supplement 1, "Inadvertent ECCS Actuation at Power," dated October 28, 1994. ADAMS Accession No. ML050320117 [pages 9-15 of file].

99. **Westinghouse 2000:** Westinghouse, NSAL-00-013, "CVCS Modeling Assumption for Loss of Offsite Power Analyses," dated August 23, 2000. ADAMS Accession No. ML103200150 [pages 131-139 of file].

100. **Westinghouse 2007:** Westinghouse, NSAL-07-10, "Loss-of-Normal Feedwater Loss-of-Offsite AC Power Analysis PORV Modeling Assumptions," dated November 7, 2007. ADAMS Accession No. ML100140163 [pages 23-26 of file].

101. **Westinghouse 2011:** Westinghouse, "AP1000 Design Control Document," Revision 19, dated June 13, 2011. ADAMS Accession No. ML11171A500.

102. **WOG 1982:** Westinghouse Owners Group (WOG), letter from Oliver D. Kingsley, Alabama Power Company, to Harold R. Denton, U.S. NRC, "NUREG-0737, Item II.D.1, 'Pressurizer Safety Valve Operability,'" dated July 27, 1982. Forwards Westinghouse WCAP-10105, "Review of Pressurizer Safety Valve Performance as Observed in the EPRI Safety and Relief Valve Test Program," dated June 1982. [This document is not in ADAMS, but is available through the NRC Public Document Room using Accession No. 8208190307 and microfiche location 14387:189-301.]

APPENDIX E: LIST OF ABBREVIATIONS

Comment [CT]: I'll work on this next, but I don't expect it to need much review by others.

From:	Spencer, Michael
To:	Scarbrough, Thomas; Clark, Theresa
Cc:	Holahan, Gary; West, Steven
Subject:	RE: 2016 Backfit Panel Tom discussion of NRR issues 2016 08 18 Gary MAS.docx
Date:	Friday, August 19, 2016 8:52:54 AM

Tom thanks for your responses to my comments. I am also fine with your minor changes.

From: Scarbrough, Thomas
Sent: Thursday, August 18, 2016 7:39 PM
To: Clark, Theresa <Theresa.Clark@nrc.gov>
Cc: Spencer, Michael <Michael.Spencer@nrc.gov>; Holahan, Gary <Gary.Holahan@nrc.gov>; West, Steven <Steven.West@nrc.gov>
Subject: RE: 2016 Backfit Panel Tom discussion of NRR issues 2016 08 18 Gary MAS.docx

Theresa,

I am fine with the edits by Gary and Michael.

Attached is the Gary/Michael version including my responses to Michael's comments and applicable edits.

This version is also in my Tom – Working folder.

Thanks.
Tom

From: Spencer, Michael
Sent: Thursday, August 18, 2016 6:31 PM
To: Holahan, Gary <Gary.Holahan@nrc.gov>; West, Steven <Steven.West@nrc.gov>; Scarbrough, Thomas <Thomas.Scarbrough@nrc.gov>; Clark, Theresa <Theresa.Clark@nrc.gov>
Subject: 2016 Backfit Panel Tom discussion of NRR issues 2016 08 18 Gary MAS.docx

My comments on top of Gary's.

From:	Spencer, Michael
To:	Clark, Theresa
Cc:	Holahan, Gary; West, Steven; Scarbrough, Thomas
Subject:	RE: ACTION: package for panel report
Date:	Tuesday, August 23, 2016 12:05:09 PM

Theresa, I looked at the updated documents.

On the report: Page 5 has a "that that."

The cover memo (page 2) has the following reference to the NRR panel member that I thought we were going to get rid of: "The Panel notes, as did a member of the earlier NRR backfit appeal panel (ADAMS Accession No. ML16081A405), that the issue of pressurizer valve performance following water discharge appears to have generic applicability, and is not specific to Byron and Braidwood."

I have no other comments, but I will continue the typo check of the references (haven't found any yet).

Michael

From: Clark, Theresa
Sent: Tuesday, August 23, 2016 11:50 AM
To: Sprogeris, Patricia <Patricia.Sprogeris@nrc.gov>
Cc: Holahan, Gary <Gary.Holahan@nrc.gov>; West, Steven <Steven.West@nrc.gov>; Scarbrough, Thomas <Thomas.Scarbrough@nrc.gov>; Spencer, Michael <Michael.Spencer@nrc.gov>
Subject: ACTION: package for panel report
Importance: High

Patti,

Could you please create a package in ADAMS and add the two attached documents, then prepare a paper concurrence package? (It does not need to be pinked, just printed and assembled.) Feel free to fix formatting (e.g., margin for letterhead) but please do not trouble yourself with editing as we have already gone over it many times. Please do add the ML#s on the front cover of the report file and on the concurrence page of the memo file.

I need this sometime this afternoon if possible.

Panel folks—FYI. Prizes for anyone who finds an error at this point ☺. I will let you know once I have a package ready for signature (and we can work in any concurrence edits there).

THANKS!!

--

Theresa Valentine Clark
Executive Technical Assistant (Reactors)

U.S. Nuclear Regulatory Commission
Theresa.Clark@nrc.gov | 301-415-4048 | O-16E22

From:	Clark, Theresa
To:	West, Steven; Spencer, Michael; Holahan, Gary; Scarbrough, Thomas
Subject:	RE: Backfit Appeal Panel Report (MASTER) - 2016-08-22 R2 - MAS
Date:	Tuesday, August 23, 2016 7:41:27 AM

Thanks—I put them all in (except for the things that your comments said you would send separately).

From: West, Steven
Sent: Tuesday, August 23, 2016 7:34 AM
To: Spencer, Michael <Michael.Spencer@nrc.gov>; Holahan, Gary <Gary.Holahan@nrc.gov>; Scarbrough, Thomas <Thomas.Scarbrough@nrc.gov>; Clark, Theresa <Theresa.Clark@nrc.gov>
Subject: RE: Backfit Appeal Panel Report (MASTER) - 2016-08-22 R2 - MAS

Theresa,

In this version, I've responded to some of the previous comments and made a substantive (and I think simplifying) change to our response to Question 4.

Steve

Steven West, Deputy Director
Office of Nuclear Security and Incident Response
U.S. Nuclear Regulatory Commission

301-287-3734
Steven.West@nrc.gov

From: Spencer, Michael
Sent: Monday, August 22, 2016 6:19 PM
To: Holahan, Gary <Gary.Holahan@nrc.gov>; West, Steven <Steven.West@nrc.gov>; Scarbrough, Thomas <Thomas.Scarbrough@nrc.gov>; Clark, Theresa <Theresa.Clark@nrc.gov>
Subject: Backfit Appeal Panel Report (MASTER) - 2016-08-22 R2 - MAS

All, attached are my comments on the document Theresa emailed out 20 minutes ago. I incorporated all of Steve's/Tom's/Theresa's edits, so any edits in the attached are mine.

Michael

From: West, Steven
To: Spencer, Michael; Holahan, Gary; Scarbrough, Thomas; Clark, Theresa
Subject: RE: Backfit Appeal Panel Report (MASTER) - 2016-08-22 R2 - MAS
Date: Tuesday, August 23, 2016 7:33:59 AM
Attachments: Backfit Appeal Panel Report (MASTER) - 2016-08-22 R2 - MAS Steve"s comments.docx

Theresa,

In this version, I've responded to some of the previous comments and made a substantive (and I think simplifying) change to our response to Question 4.

Steve

Steven West, Deputy Director
Office of Nuclear Security and Incident Response
U.S. Nuclear Regulatory Commission

301-287-3734
Steven.West@nrc.gov

From: Spencer, Michael
Sent: Monday, August 22, 2016 6:19 PM
To: Holahan, Gary <Gary.Holahan@nrc.gov>; West, Steven <Steven.West@nrc.gov>; Scarbrough, Thomas <Thomas.Scarbrough@nrc.gov>; Clark, Theresa <Theresa.Clark@nrc.gov>
Subject: Backfit Appeal Panel Report (MASTER) - 2016-08-22 R2 - MAS

All, attached are my comments on the document Theresa emailed out 20 minutes ago. I incorporated all of Steve's/Tom's/Theresa's edits, so any edits in the attached are mine.

Michael

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

Theresa V. Clark

August XX, 2016

ADAMS Accession No. MLXXXXXXXXX

TABLE OF CONTENTS

1 BACKGROUND

On June 22, 2016,[1] in accordance with NRC Management Directive (MD) 8.4,[2] the NRC Executive Director for Operations (EDO) established a Backfit Appeal Review Panel (Panel) to review the appeal by Exelon Generation Company, LLC (Exelon or the licensee) of the U.S. Nuclear Regulatory Commission (NRC) staff's determination that a backfit is necessary at Byron Station, Units 1 and 2 (Byron) and Braidwood Station, Units 1 and 2, as well as the NRC staff's application of the compliance backfit exception provided in Title 10 of the *Code of Federal Regulations* (10 CFR), Section 50.109, "Backfitting."

This backfit determination is documented in an October 9, 2015, letter (referred to as the Backfit Letter).[3] The letter describes the NRC staff's review of licensing basis documents for Byron and Braidwood. The NRC staff determined that Byron and Braidwood were not in compliance with the plant-specific design bases and several NRC regulations:

- General Design Criterion (GDC) 15, "Reactor coolant system design," in 10 CFR Part 50, Appendix A, "General Design Criteria for Nuclear Power Plants"
- GDC 21, "Protection system reliability and testability"
- GDC 29, "Protection against anticipated operational occurrences"
- Paragraph (b) of 10 CFR 50.34, "Contents of applications; technical information"

Specifically, the NRC staff determined that Byron and Braidwood do not comply with provisions in American Nuclear Society (ANS) Standard 51.1/N18.2-1973[4] for ensuring that ANS Condition II events[5] do not progress to more serious ANS Condition III events following water discharge[6] through certain valves. The NRC staff acknowledged that the NRC staff position differed from a previous staff position documented in a May 4, 2001, safety evaluation (SE) supporting a stretch power uprate (referred to as the Uprate SE).[7] However, the NRC staff determined that the backfitting was justified under the compliance exception in 10 CFR 50.109(a)(4)(i). The NRC staff directed the licensee to take action to resolve the non-compliance.

On December 8, 2015, the licensee appealed the NRC staff's decision to the Director of the Office of Nuclear Reactor Regulation (NRR), stating its disagreement with the NRC's conclusion that the compliance exception to the backfit rule applied in this case, while noting that the NRC staff had twice approved the underlying analysis.[8] The approvals referenced by the licensee

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[1] NRC 2016e (Author and year citations in footnotes refer to the designation of references in Appendix D to this report.)
[2] NRC 2013
[3] NRC 2015b – referred to as the Backfit Letter in the remainder of the report
[4] ANS 1973
[5] Specifically, inadvertent operation of the emergency core cooling system, malfunction of the chemical and volume control system, and inadvertent opening of a pressurizer safety or relief valve.
[6] For consistency in this report, the Panel uses the phrase "water discharge" rather than "water relief" or "liquid discharge" (except in direct quotes), as this is the phrase used in the Westinghouse documents that raised the issue addressed in this report.
[7] NRC 2001b – referred to as the Uprate SE in the remainder of the report
[8] Exelon 2015 – referred to as the NRR Appeal in the remainder of the report

were an August 26, 2004, license amendment associated with pressurizer safety valve (PSV) setpoints[9] and the above-referenced Uprate SE. In a letter dated May 3, 2016, the NRC responded to the licensee's appeal and reaffirmed its decision that the backfit per the compliance exception provisions of 10 CFR 50.109(a)(4)(i) is appropriate.[10]

On June 2, 2016, the licensee again appealed the NRC staff's decision, this time to the EDO.[11] The purpose of this report by the Backfit Appeal Review Panel is to provide information and recommendations to support the EDO's decision on the appeal.

1.1 Conduct of the Panel's Review

In order to establish a technically sound, well informed, and legally defensible basis for its recommendations, the Backfit Appeal Review Panel undertook a review of the relevant documents in this case. This included the licensee and NRC staff letters mentioned above; the Uprate SE and the Setpoint SE; and a June 16, 2016, letter from the Nuclear Energy Institute (NEI)[12] supporting the EDO Appeal. The Panel also reviewed many other related documents, which fall into five broad categories:

- The Backfit Rule (10 CFR 50.109), related court actions, and Commission and staff guidance on application of the Backfit Rule
- Docketed communications for Byron and Braidwood from 1982 to the present, including license amendment requests (LARs) by the licensee, NRC-issued license amendments, NRC requests for additional information (RAIs), licensee responses, meeting summaries, NRC SEs, and the licensee's Updated Final Safety Analysis Report (UFSAR)[13]
- NRC guidance relevant to the analysis of inadvertent operation of the emergency core cooling system (IOECCS) events over the period of 1981 to the present, including Standard Review Plan (SRP) Section 15.0, Sections 15.5.1 – 15.5.2, and Section 15.6.1[14]
- Westinghouse Nuclear Safety Advisory Letter (NSAL) 93-013[15] and its Supplement 1[16], as well as docketed communications regarding actions taken by other licensees in response to Westinghouse NSAL-93-013
- The history of NRC and industry activities related to power operated relief valves (PORVs), their block valves, and PSVs (including Three Mile Island (TMI) Action Plan Items II.D.1, II.D.3, II.G.1, and II.K.3 as documented in NUREG-0737[17], as well as

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[9] NRC 2004b – referred to as the Setpoint SE in the remainder of the report
[10] NRC 2016d – referred to as NRR Appeal Decision in the remainder of the report
[11] Exelon 2016a – referred to as EDO Appeal in the remainder of the report
[12] NEI 2016
[13] Exelon 2002 and Exelon 2014 (The Panel reviewed other revisions as well, but they are not included in Appendix D as they are not referenced in this report.)
[14] NRC 1981a, NRC 1981b, NRC 1981c, NRC 2007a, NRC 2007b, and NRC 2007c
[15] Westinghouse 1993
[16] Westinghouse 1994
[17] NRC 1980c – referred to as the TMI Action Plan in the remainder of the report; lessons learned from TMI were also presented in NUREG-0578 (NRC 1979a), NUREG-0585 (NRC 1979b), and NUREG-0660 (NRC 1980a)

Generic Letter 89-10[18] and its supplements), Electric Power Research Institute (EPRI) valve testing, and operating experience (NUREG/CR-7037[19])

In addition to the document review, the Panel had the benefit of meetings with NRR (both the Division of Safety Systems and the Division of Engineering), the Office of the General Counsel, and the NRC Committee to Review Generic Requirements (CRGR). Both Exelon (Bradley Fewell, Senior Vice President of Regulatory Affairs) and NEI (Tony Pietrangelo, Senior Vice President and Chief Nuclear Officer) declined offers for a public meeting, but indicated a willingness to provide information if the Panel identified the need. The Panel did not identify a need for additional information from either Exelon or NEI to complete the review documented in this report.

At the request of the Panel, the Office of Nuclear Regulatory Research (RES) conducted risk analyses using the NRC's Standardized Plant Analysis Risk model for Byron Unit 1.[20] These analyses informed the Panel's response to the question from the EDO regarding the risk significance of the relevant accident sequences.

Given that the Backfit Rule creates a structured process for changes to previous NRC staff positions—in effect, placing the burden of proof on the NRC staff—the Panel determined that this level of historical review and staff interaction was necessary to provide context for consideration of the validity of the backfit.

Comment [CT]: I'm open to comment or deletion on this, but this is my attempt to address Steve's comment about the first (NRR) panel review scope.

Comment [SW]: I'd keep it and I have a few edits to provide separately.

1.2 Proposed Compliance Backfit and Exelon Appeals

In the Backfit Letter, the NRC staff informed Exelon that it had determined that Byron and Braidwood are not in compliance with GDCs 15, 21, and 29; 10 CFR 50.34(b); and the plant-specific design bases that were expected to demonstrate there will be no progression of ANS Condition II events to ANS Condition III events. The NRC staff stated that based on its review of Byron and Braidwood UFSAR Sections 15.5.1, 15.5.2, and 15.6.1, the UFSAR predicts water discharge through a valve that is not "qualified" for water discharge. Therefore, the NRC staff concluded that the UFSAR does not contain analyses that demonstrate that the plants' structures, systems, and components (SSCs) meet the design criteria for ANS Condition II events as stated in Byron and Braidwood UFSAR Section 15.0.1.2. Based on the SE attached to its letter,[21] the NRC staff found that the licensee must take action to resolve the non-compliance.

The Backfit SE addressed three accident analyses in Chapter 15 of the Byron and Braidwood UFSAR: (1) IOECCS; (2) chemical and volume control system (CVCS) malfunction that increases reactor coolant inventory; and (3) inadvertent opening of a pressurizer safety or relief valve (IOPORV). The NRC staff noted that each ANS Condition II event must be shown to meet the following:

1. no fuel damage,

2. no overpressure of the reactor coolant system (RCS) or main steam system, and

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[18] NRC 1989
[19] NRC 2011
[20] NRC 2016f
[21] Referred to as the Backfit SE in the remainder of the report.

3. no progression into an event of a more serious category without another independent fault.

Regarding an IOECCS, the NRC staff stated in Section 3.1.2.1 of the Backfit SE that use of the block valve to isolate a stuck-open PORV was unacceptable. The NRC staff stated that Westinghouse recommended this approach in 1993, and that the NRC staff rejected this approach in 2005 (RIS 2005-29[22]).

In Section 3.1.2.4 of the Backfit SE, the NRC staff stated that the Byron and Braidwood IOECCS analysis depended on water discharge through the PSVs. The NRC staff faulted the licensee for "not appl[ying] the single-failure assumption" and stated that the following information was necessary to support water qualification of the PSVs:

1. In accordance with the American Society of Mechanical Engineers (ASME) *Boiler and Pressure Vessel Code* (BPV Code), Section III, provide the original Overpressure Protection Report defining operating conditions and required relief capacities, and manufacturer's certification and test results

2. In accordance with the ASME *Code for Operation and Maintenance of Nuclear Power Plants* (OM Code), provide inservice test history for PSVs, including water and steam tests, or provide correlation test for alternative test fluid.

Regarding a CVCS malfunction, the NRC staff stated in Section 3.2 of the Backfit SE that the licensee had not provided an analysis for the CVCS malfunction that increases reactor coolant inventory that demonstrated the plants' ability to meet the requirements of an ANS Condition II event.

Regarding an IOPORV, the NRC staff stated in Section 3.3 of the Backfit SE that the licensee had not provided an analysis for the IOPORV that extends long enough into the transient to demonstrate the event would not transition from an ANS Condition II event to an ANS Condition III event.

In the Backfit SE, the NRC staff referenced Millstone[23] and Callaway[24] license amendments as examples of licensees upgrading PORVs for water discharge; a Beaver Valley extended power uprate (EPU) license amendment[25] as an example of qualifying PORVs for water discharge; and Turkey Point[26] and St. Lucie Unit 2[27] EPU amendments as additional precedent in support of the backfit decision.

In the NRR Appeal, Exelon asserted that the NRC had not justified invoking the compliance exception to the backfit rule. Exelon stated that the NRC approved its IOECCS analysis in both the Uprate SE and the Setpoint SE.

In the NRR Appeal Decision, the NRC staff stated that the previous NRC approvals in 2001 and 2004 were inconsistent with the Agency's general position on the known and established standard at issue—in this case, the progression of ANS Condition II events to higher level

Comment [CT]: **STEVE:** I didn't insert the "had" as I've taken them out a lot of places (and it doesn't say "had rejected" later in the sentence). But if you really want it I can put it in. Same in the "In the NRR Appeal" paragraph at the bottom of the page.

Comment [SW]: I'm fine with the way you have it.

Comment [CT]: Comment placeholder – think about more discussion on scope of NRR appeal.

Comment [MAS]: The NRR appeal decision encompassed both 2001 and 2004:

"The NRC erred in approving a sequence of events that allowed the inadvertent operation of the emergency core cooling system, chemical and volume control system malfunction, and inadvertent opening of a pressurizer safety or relief valve analyses in the 2001 and 2004 Safety Evaluations (ADAMS Accession Nos. ML011420274 and ML042250516, respectively) to credit water relief through pressurizer safety valves (PSVs) that were not water qualified. The NRC has consistently applied the prohibition of progression of Condition II events, and the 2001 and 2004 approvals occurred because the NRC staff understood the PSVs to be qualified for water relief when, in fact, they were not."

Comment [SW]: I have some suggested revisions that I have marked in a different version of this report and will provide separately.

[22] NRC 2005b
[23] NRC 1998
[24] NRC 2000
[25] NRC 2006
[26] NRC 2012a
[27] NRC 2012b

events. The NRC staff stated that the fact that the NRC staff were aware of references to EPRI reports on the ability of these non-water qualified PSVs to reseat in certain circumstances was not sufficient to support the licensee's position on the compliance backfit.

In the EDO Appeal, Exelon stated that the NRC had misidentified the "known and established standard" at issue as the prohibition of ANS Condition II events progressing to ANS Condition III events. Exelon asserted that the standard in question concerns what is necessary to "qualify" valves for water discharge. Exelon contended that this standard was the EPRI testing and analysis, and that the NRC agreed that Byron and Braidwood met this standard. Exelon also contended that the change in NRC staff position on prior approvals was not a mistake of fact, but rather a new or modified interpretation of compliance with NRC requirements, for which use of the compliance exception provided for in the Backfit Rule was not appropriate.

1.3 Backfit Rule and the Compliance Exception

Backfitting is defined by 10 CFR 50.109(a) as:

> … the modification of or addition to systems, structures, components, or design of a facility; or the design approval or manufacturing license for a facility; or the procedures or organization required to design, construct or operate a facility; any of which may result from a new or amended provision in the Commission's regulations or the imposition of a regulatory staff position interpreting the Commission's regulations that is either new or different from a previously applicable staff position …

Comment [CT]: **STEVE:** Michael put this period in, so I figured it was a legal convention. Keeping it for now.

Comment [MAS]: This a legal convention, but since this is not a legal document, it is OK by me if you want to delete the fourth dot. If so, this should be done globally.

Comment [SW]: When I first saw it, I saw a typo, but I agree that it works. I'd keep the dot.

Unless one of three specified exceptions apply, the NRC may impose a backfit only if it performs a backfit analysis in accordance with 10 CFR 50.109(a)(2) and determines in accordance with 10 CFR 50.109(a)(3) "that there is a substantial increase in the overall protection of the public health and safety or the common defense and security to be derived from the backfit and that the direct and indirect costs of implementation for that facility are justified in view of this increased protection."

Section 50.109(a)(4) sets forth the three exceptions to the requirements of 10 CFR 50.109(a)(2) and (a)(3). The first exception, the compliance exception, applies if the "modification is necessary to bring a facility into compliance with a license or the rules or orders of the Commission, or into conformance with written commitments by the licensee." The second and third exceptions relate to actions necessary to ensure adequate protection or to actions that involve defining or redefining adequate protection.

The Commission explained its intended application of the compliance exception in the Statements of Consideration (SOC) accompanying the 1985 final rule amending 10 CFR 50.109:[28]

> The compliance exception is intended to address situations in which the licensee has failed to meet known and established standards of the Commission because of omission or mistake of fact. It should be noted that new or modified interpretations of what constitutes compliance would not fall within the exception and would require a backfit analysis and application of the standard.

[28] NRC 1985, at 38103

In the same SOC, the Commission acknowledged that staff interpretations of rules are not legally binding, but the Commission also stated that "staff interpretations of broadly stated rules are often necessary to give a rule effect and in some instances may be a causal factor in initiating a backfit."[29]

By its terms, the compliance exception applies to actions necessary for compliance with rules, licenses, and orders, or for conformance with written commitments.[30] Also, the Commission explicitly acknowledged the importance of staff interpretations of rules in the regulatory process. Thus, the Panel understands the term "known and established standard" to include standards established in rules, licenses, orders, and written commitments, and NRC interpretations of rules. Some standards may be broad-based, while others may apply only to a limited number of plants. As stated in NUREG-1409, "[i]nformal or formal communications to one licensee are not official positions to all licensees. ... Orders, licenses, and written commitments are applicable only to a particular licensee."

The failure to meet a known and established standard is grounds for a compliance backfit if this failure is due to "omission or mistake of fact." Thus, if a licensee obtains NRC approval of an alternative to a specific standard set forth in guidance, that standard and guidance could not be used to support a compliance backfit unless the NRC's approval of the alternative was based on an omission or mistake of fact. "Known and established standards" are to be distinguished from "new or modified interpretations of what constitutes compliance," which do not fall within the compliance exception. The Panel understands the term "new or modified interpretations" to include situations where the NRC staff has, in effect, "changed its mind" on how to interpret the language of a requirement or on how much assurance is necessary to conclude that the requirement is met. Levels of assurance might be established in terms such as acceptable probabilities or consequences, conservative assumptions, or sufficient margin.

Additional background information on the Backfit Rule and the compliance exception is provided in Appendix A to this report.

1.4 A Brief History of Pressurizer Valve Issues

Appendix B to this report provides a summary of the NRC and industry's testing, evaluation, and other consideration of PORVs and PSVs since the TMI Unit 2 (TMI-2) accident in 1979. This historical review provides context for discussion of valve "qualification" in the Backfit SE. It also provides the basis for the Panel's conclusions regarding the "known and established standard" for "qualification" in the context of TMI Action Plan Item II.D.1 and subsequent activities, as well as how it should be interpreted in the Byron and Braidwood licensing basis.

In light of the NRC staff's assertion that the licensee had not applied the "single-failure assumption" as noted above, the Panel also considered the applicability of the single failure

[29] NRC 1985, at 38102. The 1985 backfit rule was vacated by a Federal court on grounds unrelated to the compliance backfit exception. See *Union of Concerned Scientists v. U.S. Nuclear Regulatory Com'n*, 824 F.2d 108, 119-20 (1987). In 1988, the Commission amended the backfit rule (NRC 1988b) to address the court's concerns, but did not change the 1985 rule's compliance exception provision. Thus, the quoted statements from the 1985 rule are the applicable expression of Commission intent regarding compliance backfits.

[30] NUREG-1409 (NRC 1990c) defines written commitments broadly to include the "final safety analysis report, licensee event reports, and docketed correspondence, including responses to NRC bulletins, generic letters, inspection reports, or notices of violation and confirmatory action letters."

criterion to PSVs. The Panel expended considerable effort in searching for an answer to what appears to be a simple question: "Are PSVs active components subject to the single failure criterion, or are they passive components exempt from the single failure criterion?" NRR staff have taken the position that PSVs have consistently been treated as active components.

Comment [CT]: Differing views on this topic... but as I am of the "a historical" persuasion personally, I'm taking Tom's edit ☺.

Comment [SW]: I support.

In the Panel's evaluation of the treatment of PSV failure potential (Section 3 below), a historical perspective is provided. In general, the Panel found that the classification of a component as "active" or "passive" depends on its design, application, and function. For example, passive components almost always do not need external power; usually do not need an external actuator (e.g., signal)[31]; sometimes do not involve any mechanical motion (e.g., movement of a valve disc)[32]; and sometimes do not involve any motion, either fluid or mechanical (e.g., piping). While it does not represent formal NRC guidance, additional views on passive components are included in International Atomic Energy Agency (IAEA) TECDOC-1624.[33] This document states that "[s]afety related terms such as passive and inherent safety have been widely used, particularly with respect to advanced nuclear plants, generally without definition and sometimes with definitions inconsistent with each other." This guidance further defines four levels of "passivity" to "help eliminate confusion and misuse of the terms by members of the nuclear community." In addition, SECY-05-0138[34] also acknowledged and discussed inconsistencies in the use and application of the term "passive." Additional consideration of this topic by the Panel is documented in Section 3.10 below.

The introduction to the GDCs and the related footnote define the applicability of the single failure criterion in terms of electrical versus fluid systems, and active versus passive components. Neither the GDCs nor NRC guidance define which characteristics of passive components are necessary to make a component exempt from the single failure criterion. Some examples are clear: pipes are passive components and pumps and motor-operated valves that operate to perform their safety functions are active components. As discussed in Section 3.6 below, check valves might be classified as active or passive components depending on certain specific considerations.

With respect to PSVs, the ASME BPV Code applicable to Byron and Braidwood includes requirements for overpressure protection that relate to the single failure criterion through several specific design and construction requirements. As a result, the PSVs are conservatively sized with sufficient margin to accommodate a single failure although the single failure criterion is almost never explicitly discussed or applied in accident analyses. The Byron and Braidwood UFSAR states that "adequate overpressurization protection is provided by the three installed safety valves." Neither the UFSAR system descriptions nor the safety analyses provide detailed discussions of potential PSV failures or their consequences. The principal discussion of

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[31] For example, SECY-77-439 (NRC 1977) states: "Examples [of passive failures in fluid systems] include the failure of a simple check valve to move to its correct position when required, the leakage of fluid from failed components, such as pipes and valves—particularly through a failed seal at a valve or pump—or line blockage. Motor-operated valves which have the source of power locked out are allowed to be treated as passive components."

[32] For example, NUREG-1800 (NRC 2001c) states that "'[p]assive' structures and components, for the purpose of the license renewal rule, are those that perform an intended function … without moving parts or without a change in configuration or properties … 'passive' may also be interpreted to include structures and components that do not display 'a change of state.'"

[33] IAEA 2009

[34] NRC 2005a

potential PSV failures in the accident analyses occurs in the evaluation of an inadvertent opening of a PSV in UFSAR Section 15.6.1.

Most relevant for the current issue, the Byron and Braidwood UFSAR analyses of overpressure events (e.g., loss of load, loss of feedwater) do not apply the single failure criterion to cause a PSV to stick open (i.e., fail to reseat) when opening on steam flow. In addition, the UFSAR Feedwater System Pipe Break analysis (Chapter 15.2.8) does not apply the single failure criterion to cause a PSV to stick open either during steam discharge or during water discharge. A survey of other Westinghouse-designed plants showed that this treatment of PSV valve performance during anticipated operational occurrences (AOOs, similar to ANS Condition II events) and postulated accidents (similar to ANS Condition IV events) has been consistent and without any identified exceptions.[35]

1.5 History and Review of Westinghouse NSAL and Related Activities

Appendix C to this report provides the Panel's review of the issues identified by Westinghouse in NSAL-93-13 and its Supplement 1, how various licensees responded to these issues, and how the NRC was involved in reviewing and approving these actions. This review provides the basis for the Panel's conclusions related to the approach taken by Byron and Braidwood to address these issues in their licensing basis, as well as on the "known and established standard" for event escalation from ANS Condition II to ANS Condition III, referred to hereafter as the "non-escalation position."

2 SUMMARY OF THE APPEAL REVIEW PANEL FINDINGS

For the reasons provided in Section 3, the Panel concluded that in 2001 and 2004 and at present, the known and established standard of the Commission is 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. The Panel also concluded that, in preparing the Uprate SE and the Setpoint SE, the NRC staff exercised reasonable and well-informed engineering judgment when the NRC staff concluded that the PSVs were unlikely to stick open. The non-escalation position does not establish specific standards for valve qualification, so the non-escalation position, standing alone, provides no basis for rejecting the licensee's reliance on EPRI valve testing. Moreover, the Panel found that no mistake or error occurred in the licensee's or previous staff's reliance on the EPRI testing program that included an evaluation of water discharge through pressurizer valves.[36] Therefore, the Panel also concluded that the NRC staff's position on valve qualification in the Backfit SE is a new or modified interpretation of what constitutes compliance.

The Panel also concluded that the issue of pressurizer valve performance following water discharge appears to have generic applicability, and is not specific to only Byron and Braidwood. The Panel believes that resolution of this issue would have benefited from consideration of the generic nature of the issue through the appropriate NRC processes. The Panel included additional information about this finding in Section 6 and Appendices B and C below.

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[35] Examples include Watts Bar (NRC 1982 and TVA 1983), North Anna (NRC 1976), and AP1000 (Westinghouse 2011).

[36] "Pressurizer valves" is used in this report to refer to either PORVs or PSVs when discussing issues common to both types of valves.

3 DISCUSSION

The compliance exception to the Backfit Rule is intended to address failures to meet known and established Commission standards because of omission or mistake of fact. New or modified interpretations of what constitutes compliance do not fall within the exception. The Panel reviewed and evaluated the information referenced in this report to determine if, in 2001 and 2004, there was a known and established standard of the Commission relating to the potential for PSVs to fail following water discharge during IOECCS events.

In addition, the Panel considered the issue of "known and established standards of the Commission" as it relates to "event escalation." The NRR Appeal Decision stated that the Backfit SE "showed that the approvals at issue for Braidwood and Byron were inconsistent with the Agency's general position on the known and established standard at issue, in this case the progression of [ANS] Condition II events." The Panel recognizes that the non-escalation position, although not included in NRC regulations, is widely referenced in reactor licensing bases as an approach for addressing AOOs and postulated accidents as articulated in the GDCs. The non-escalation position is incorporated in Section 15.0.1.2 of the Byron and Braidwood UFSAR as "By definition, these faults (or events) do not propagate to cause a more serious fault, i.e., [ANS] Condition III or IV events."

Exelon and the Panel agree that the non-escalation position is now, and was in 2001 and 2004, a part of the licensing basis of both Byron and Braidwood. In addition, the Panel supports the NRC staff's view that non-escalation (from ANS Condition II to ANS Condition III or IV) is a known and established standard applicable to Byron and Braidwood. However, the Panel also agrees with Exelon that the fundamental issue is not the non-escalation position, as the NRC staff contends, but rather the appropriate standard for PSV water discharge. In the absence of a PSV failure to reseat, the concerns articulated by the NRC staff in the backfit related to event classification, event escalation, and compliance with 10 CFR 50.34(b) and GDCs 15, 21, and 29 would no longer be at issue.

The Panel's evaluation of the treatment of PSV failure potential includes an assessment of multiple relevant references, which are discussed chronologically in the sections that follow.

3.1 General Design Criteria (1971)

In 1971, the Atomic Energy Commission published the GDCs, which had been under development since 1965.[37] The introduction to 10 CFR Part 50, Appendix A addresses "Single Failure" in the section on Definitions and Explanations. The paragraph on single failures includes a footnote stating: "The conditions under which a single failure of a passive component in a fluid system should be considered in designing the system against a single failure are <u>under development</u>" (emphasis added).

3.2 Commission Paper on Single Failure (1977)

In response to several staff concerns and differing views on the subject of application of the single failure criterion, the Acting Director of NRR issued SECY-77-439 "[t]o inform the Commission of the present status and future use of the Single Failure Criterion as a tool in the reactor safety process."[38] In part, that paper addressed the application of the single failure

[37] AEC 1971
[38] NRC 1977

criterion to passive components in fluid systems, stating that "[a]pplication of the [single failure] concept is complicated by the interrelationships between the various fluid and electrical systems and their supporting auxiliaries in a nuclear power plant. Furthermore, there is a need to stipulate the events and associated assumptions which must be considered during application of the Single Failure Criterion."

SECY-77-439 specifically spoke to how "additional passive failures"—that is, failures in addition to the initiating event—had been and should be addressed, stating (with emphases added):

> During subsequent years [since the single failure footnote quoted above was published] staff <u>assumptions regarding the nature of passive failures which should be considered have not been completely consistent</u> and there has been some disagreement. However, on the basis of the licensing review experience accumulated in the period since 1969, it has been judged <u>in most instances that the probability of most types of passive failures in fluid systems is sufficiently small that they need not be assumed</u> in addition to the initiating failure in application of the Single Failure Criterion to assure safety of a nuclear power plant.

Furthermore, SECY-77-439 provides definitions and examples for distinguishing between active and passive failures. Among these examples, SECY-77-439 cites "the failure of a simple check valve to move to its correct position when required" as a passive failure. Of the examples cited in SECY-77-439, the check valve example is most similar from a mechanical perspective to the PSV failure addressed in the Backfit SE, as explained below in the discussion of SECY-94-084.

SECY-77-439 also stresses the use of engineering judgment relating to the probability of component failure and does not suggest that valve "certification" or "qualification" in accordance with ASME standards should be invoked as the basis for such decisions.

3.3 TMI Action Plan Item II.D.1 (1980)

As an element of the TMI Action Plan, the NRC staff required licensees to address the capability of relief and safety valves to perform their intended functions without failure. Specifically, Item II.D.1 states that "[p]ressurized-water reactor [PWR] and boiling-water reactor [BWR] licensees and applicants shall conduct testing to qualify the [RCS] relief and safety valves under expected operating conditions for design-basis transients and accidents." With reference to planned EPRI testing and other generic industry test programs, NUREG-0737 specified provisions for then-operating nuclear power plants and applicants for operating licenses and holders of construction permits to address the TMI Action Plan items, including Item II.D.1. NUREG-0737 stated, for the performance testing of relief and safety valves for Item II.D.1, that "[t]he testing should demonstrate that the valves will open and reclose under the expected flow conditions."

Although limited in scope, the EPRI test results did not identify any generic issues with PSVs or PORVs sticking open following water discharge. The NRC staff approvals summarized below show that the word "qualify" in this TMI Action Plan item was not intended to refer to ASME valve certification or qualification. Instead, "qualify" was used in a less formal sense to refer to a reasonable judgment that the valve would open to relieve pressure and then reliably reseat. As referenced in NUREG-0737, the EPRI test program was the widely used approach to address TMI Action Plan Item II.D.1 at PWR nuclear power plants. The Westinghouse Owners Group submitted WCAP-10105 to the NRC in 1982 to demonstrate the acceptability of the EPRI testing program for PSVs and PORVs in Westinghouse-designed PWRs.[39]

3.4 NRC Closure of TMI Action Plan Item II.D.1 for Byron and Braidwood (1988-1990)

A 1988 letter from the NRC staff to the licensee for Byron found the licensee's reliance on EPRI testing of PSVs to be acceptable.[40] The 1988 SE states that the test program was designed "[t]o reconfirm the integrity of the overpressure protection system and thereby assure that the [GDCs] are met." As discussed in Appendix B to this report, the 1988 SE described the NRC staff's evaluation of the PSVs and PORVs for feedwater line break accidents that would include water discharge, and determined that the EPRI tests were applicable to the Byron and Braidwood PSVs and PORVs. Based on the NRC staff and contractor review, the 1988 SE found that the performance of the PSVs and PORVs was acceptable based on the EPRI tests.

For the specific extended high pressure injection event, the 1988 SE states that water discharge through the PSVs and PORVs could be disregarded because of the long time available for operator action. However, the SE addressed water discharge through the PSVs and PORVs as part of the feedwater line break evaluation.

In the cover letter for the 1988 SE, the NRC staff states that the licensee should develop and adopt plant procedures to inspect the pressurizer valves after each lift involving loop seal or water discharge. The 1988 SE contains no reference to or suggestion of a need for certification of these valves in accordance with the ASME BPV Code for water discharge capability. In 1990, the NRC staff also found the use of the EPRI test program similarly acceptable for Braidwood.[41]

3.5 Westinghouse NSAL-93-013 and Supplement 1 (1993-1994)

In 1993, Westinghouse sent NSAL-93-013 to operating nuclear power plants in response to its discovery that potentially non-conservative assumptions had been used in the licensing analysis of the IOECCS event. Westinghouse recommended that licensees determine if their pressurizer safety relief valves (PSRVs)[42] "are capable of closing following discharge of subcooled water." Westinghouse noted that the PSRVs might have been designed or "qualified" to relieve subcooled water. Westinghouse also noted that "licensees may have qualified these valves in compliance to NUREG-0737, Item II.D.1." If the PSRVs were not designed or qualified for subcooled water discharge, Westinghouse recommended that licensees reevaluate the IOECCS event with three possible options of (1) reducing emergency core cooling system (ECCS) flow used in the safety analysis, (2) using a less restrictive operator response time, or (3) crediting the use of one or more PORVs to help mitigate the accident.

Later, in Supplement 1 to NSAL-93-013, Westinghouse alerted licensees to potential reduced time for operator action if a positive displacement pump (a typical component of the CVCS) were in service, and to the need to qualify the PSRVs and the piping downstream of the PSRVs and PORVs if water discharge from the pressurizer is predicted.

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[39] WOG 1982
[40] NRC 1988c, referred to as the 1988 SE
[41] NRC 1990a
[42] Westinghouse used the term PSRVs. The specific valves for Byron and Braidwood should be designated as "safety valves" or "pressurizer safety valves" as they are by the manufacturer, in the ASME BPV Code, and by the licensee. This difference in terminology is not significant to any of the findings or conclusions in this report.

Some licensees submitted license amendments that involved improvements to the PORVs and their circuitry to avoid water discharge through the PSVs (e.g., Salem[43], Millstone[44], Callaway[45], and Diablo Canyon[46]). The NRC staff review and approval of those proposed improvements relied on engineering judgment relative to the various test information and PORV circuitry upgrades described by individual licensees. The licensee for Byron and Braidwood submitted an LAR for similar PORV improvements,[47] but that request was later withdrawn.[48]

As indicated below, the Panel's sampling review found at least two plants, in addition to Byron and Braidwood, that chose to address this issue by crediting the capability of PSVs to relieve water, based on the EPRI testing performed in response to TMI Action Plan Item II.D.1.

3.6 Commission Paper on Passive Plant Designs (1994)

In 1994, in preparation for the design certification reviews of passive reactor designs (e.g., the Westinghouse Advanced Passive 1000 (AP1000) and the General Electric Economic Simplified Boiling-Water Reactor (ESBWR)), the NRC staff presented nine issues to the Commission for policy decisions.[49] Although PSV categorization and performance requirements were not explicitly addressed, the paper does include an issue on "Definition of Passive Failure" and an extensive discussion on whether check valves are passive or active components and how they should be addressed in current plants and future passive designs.

SECY-94-084 recognized the GDCs and SECY-77-439 as establishing long-standing requirements and guidance in this area. The paper acknowledged that the industry (including EPRI documents and ANSI/ANS 58.9[50]) have been inconsistent with respect to check valve failures, sometimes considering them as "active failures" and sometimes as "passive failures." In SECY-77-439, however, the NRC staff stated that the failure of a simple check valve to move to its correct position when required was a "passive failure." In addition, SECY-94-084 states that "[i]n licensing reviews, however, only on a long-term basis [e.g., long-term recirculation cooling following a loss of coolant accident (LOCA)] does the NRC staff consider passive failures in fluid systems as potential accident initiators in addition to initiating events." The paper also states that "[f]or current plants, the NRC staff normally treats check valves, except for those in containment isolation systems, as passive devices during transients or design-basis accidents."

Furthermore, SECY-94-084 states that "[r]edefining check valves as active components, subject to consideration for single active failures would cause these valves to be evaluated in a more stringent manner than that used in previous licensing reviews" (emphasis added). The NRC staff then recommended (and the Commission agreed[51]) that the NRC staff should "maintain the current licensing practice for passive component failures on the passive [advanced light water reactor] ALWR designs, and to redefine check valves, except for those whose proper function can be demonstrated and documented, in the passive safety systems as active components

Comment [CT]: A note in defense of tense...

For SERs, my informal convention had been to use present tense for documents of lasting value ("the FSAR says") and past tense for things that happened ones ("In the RAI response, the licensee said").

But, search and replace on tense is hard, and the conventions are different for a document like this, so I'm fine with past tense (or some sort of "would be" for accident type things) in this report.

I would be fine deferring to Gary's mom's rules too ☺.

Comment [SW]: I defer to Theresa's convention. It makes sense and should result in a high degree of consistency throughout.

[43] NRC 1997
[44] NRC 1998
[45] NRC 2000
[46] NRC 2004a
[47] ComEd 1998
[48] ComEd 1999
[49] NRC 1994a
[50] ANS 1981
[51] NRC 1994b

subject to single failure consideration." Therefore, the NRC's position on check valves was changed only for passive ALWR designs going forward.

The Panel considered the opening function of check valves and PSVs to be similar in that they both open through the motion of the valve disk under differential pressure with no external signal or motive power. The Panel also recognized that the ambiguity with respect to "passive" versus "active" component definitions and nomenclature exists for safety valves. In addition, the passive or active classification of check valves or safety valves may differ based on design considerations, inservice testing, or accident analyses. For example, the PSVs and PORVs, as well as numerous check valves, are classified as active components in the Byron and Braidwood inservice testing programs. However, for purposes of applying the single failure criterion in the GDC context, the Panel concluded that it is appropriate to consider the potential failure of a PSV following water discharge as a passive failure (consistent with the treatment of check valve failures for the operating fleet), provided the licensee or applicant qualifies the performance of the PSV in an acceptable manner. In the case of Byron and Braidwood, the NRC staff accepted the EPRI testing associated with TMI Action Plan Item II.D.1 to provide this qualification.

Comment [SM]: This is one edit I would like to discuss tomorrow; in particular the relationship between "passive" and qualification testing, and whether the single failure criterion is the regulatory basis for the TMI item calling for qualification.

Comment [CT]: Edited a bit from Tom's suggestion.

3.7 Draft Standard Review Plan Revision (1996)

The 1996 draft revision to SRP Sections 15.5.1 – 15.5.2 on IOECCS and CVCS malfunctions includes extensive updates to the 1981 revision, but neither version includes any discussion, criteria, or guidance on applying ASME Code requirements to PSVs or on applying the single failure criterion or any other failure assumption to PSVs.[52]

3.8 Power Uprate Reviews and License Amendments (2001-2006)

As part of the 2001 power uprate review for Byron and Braidwood, the NRC staff approved the analysis of an IOECCS (UFSAR Section 15.5.1) that included pressurizer filling, PSV water discharge, ECCS termination, and PSV closure. In the Backfit SE, the NRC staff indicated that the 2001 license amendment was predicated on the NRC's mistaken (unsubstantiated) belief that the valves were ASME-qualified (certified). However, the Panel's review of the SE and associated RAIs showed that, in 2001, the NRC staff was well aware of the nature of the EPRI testing that the licensee relied on. The Panel did not find any evidence that the licensee claimed or the NRC staff believed that the valves were "qualified" in an ASME BPV Code certification sense; rather, the record shows that the NRC staff thoroughly considered the testing conducted on valves of the type installed at the plants and applied well-informed and reasoned technical judgment in reaching its conclusion that the EPRI testing provided appropriate qualification.

The Panel confirmed its conclusions and understanding about the 2001 NRC staff review via discussions with the individual who was the responsible Section Chief in the Reactor Systems Branch at the time. He informed the Panel that the 2001 license amendment was based on the exercise of staff engineering judgment and that there was no discussion of ASME BPV Code certification or qualification of valves. In addition, the Panel found that the NRC approved power uprates for other nuclear power plants that included comparable staff evaluations of water discharge through PORVs or PSVs based on test information provided by individual licensees. For example, in 2001, the NRC granted a power uprate for Shearon Harris that included the operability of PORVs and PSVs during the discharge of subcooled water, referencing TMI Action Plan Item II.D.1.[53] As noted above, in 2006, the NRC also granted a power uprate for

[52] NRC 1996