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

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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 (Braidwood), 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 2016c (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 2001a – 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. The Director of NRR chartered a Backfit Review Panel[10] to review the licensee's appeal of the NRC staff's determination that a backfit is necessary at Byron and Braidwood, and the NRC staff's application of the compliance backfit exception. Following the NRR Backfit Review Panel's evaluation of the issue, the Director of NRR responded to the licensee's appeal in a letter dated May 3, 2016, that reaffirmed the NRC staff's decision that the backfit per the compliance exception provisions of 10 CFR 50.109(a)(4)(i) was appropriate.[11]

On June 2, 2016, the licensee again appealed the NRC staff's decision, this time to the EDO.[12] The purpose of this report 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 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)[13] 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, the transcript of a public meeting conducted during the review of the NRR Appeal,[14] and the licensee's Updated Final Safety Analysis Report (UFSAR)[15]

- 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[16]

- Westinghouse Nuclear Safety Advisory Letter (NSAL) 93-013[17] and its Supplement 1,[18] as well as docketed communications regarding actions taken by other licensees in response to Westinghouse NSAL-93-013

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[9] NRC 2004b – referred to as the Setpoint SE in the remainder of the report
[10] Referred to as the NRR Backfit Review Panel in the remainder of the report
[11] NRC 2016b – referred to as NRR Appeal Decision in the remainder of the report
[12] Exelon 2016a – referred to as EDO Appeal in the remainder of the report
[13] NEI 2016
[14] NRC 2016a
[15] 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.)
[16] NRC 1981a, NRC 1981b, NRC 1981c, NRC 2007a, NRC 2007b, and NRC 2007c
[17] Westinghouse 1993
[18] Westinghouse 1994

- 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[19], as well as Generic Letter 89-10[20] and its supplements), Electric Power Research Institute (EPRI) valve testing, and operating experience (NUREG/CR-7037[21])

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.[22] 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 establish the appropriate context for consideration of the validity of the backfit directed by the NRC staff.

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,[23] 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

[19] 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)
[20] NRC 1989
[21] NRC 2011
[22] NRC 2016d
[23] Referred to as the Backfit SE in the remainder of the report.

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 Condition 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 (Regulatory Issue Summary (RIS) 2005-29[24]).

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[25] and Callaway[26] license amendments as examples of licensees upgrading PORVs for water discharge; a Beaver Valley extended power uprate (EPU) license amendment[27] as an example of qualifying PORVs for water discharge; and Turkey Point[28] and St. Lucie Unit 2[29] EPU amendments as additional precedent in support of the backfit decision.

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[24] NRC 2005b
[25] NRC 1998
[26] NRC 2000
[27] NRC 2006
[28] NRC 2012a
[29] NRC 2012b

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.

Following the NRR Backfit Review Panel's evaluation of the NRR Appeal, the NRR Appeal Decision included a finding that the previous NRC approvals in 2001 and 2004 were not consistent with the NRC's general position on the known and established standard at issue—in this case, the ANS provisions for ensuring that ANS Condition II events do not progress to more serious ANS Condition III events. The NRR Appeal Decision also noted that the fact that the NRC staff in 2001 and 2004 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 that the compliance backfit did not apply in this case. For these reasons, the NRR Appeal Decision concluded that the backfitting directed by the NRC staff was justified under the compliance exception to the Backfit Rule.

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:[30]

> 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."[31]

By its terms, the compliance exception applies to actions necessary for compliance with rules, licenses, and orders, or for conformance with written commitments.[32] 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.

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[30] NRC 1985, at 38103

[31] 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.

[32] 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."

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 considerations 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 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.

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)[33]; sometimes do not involve any mechanical motion (e.g., movement of a valve disc)[34]; 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.[35] 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[36] 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

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[33] 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."

[34] For example, NUREG-1800 (NRC 2001b) 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.'"

[35] IAEA 2009

[36] NRC 2005a

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,” where “three” refers to the total number of PSVs installed in each unit. 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 only in the evaluation of an inadvertent opening of a PSV as an initiating event 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.[37]

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 to reclose 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

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

an evaluation of water discharge through pressurizer valves.[38] 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.

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.

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

3.1 General Design Criteria (1971)

In 1971, the Atomic Energy Commission published the GDCs, which had been under development since 1965.[39] 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."[40] 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.

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,

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[39] AEC 1971
[40] NRC 1977

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.[41]

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.[42] 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.[43]

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

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[41] WOG 1982
[42] NRC 1988c, referred to as the 1988 SE
[43] NRC 1990a

of the IOECCS event. Westinghouse recommended that licensees determine if their pressurizer safety relief valves (PSRVs)[44] "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. Westinghouse also alerted licensees 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[45], Millstone[46], Callaway[47], and Diablo Canyon[48]). 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,[49] but that request was later withdrawn.[50]

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.[51] 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

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[44] 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.
[45] NRC 1997
[46] NRC 1998
[47] NRC 2000
[48] NRC 2004a
[49] ComEd 1998
[50] ComEd 1999
[51] NRC 1994a

EPRI documents and industry standards[52]) 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[53]) 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 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 for Byron and Braidwood, the Panel concluded that it is appropriate to consider the potential failure of a PSV to reclose following water discharge as a passive failure (consistent with the treatment of check valve failures for the operating fleet), supported by the EPRI testing associated with TMI Action Plan Item II.D.1 that gave confidence in the capability of the valves.

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.[54]

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

[52] ANS 1981
[53] NRC 1994b
[54] NRC 1996

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.[55] 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 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 reclosing after water discharge.

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.

[55] NRC 2001c

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) <u>pressurizer level should not be allowed to reach a pressurizer water-solid condition.</u>".

However, the NRC staff review standard cited in the RIS (RS-001) is explicitly limited to EPU reviews and states:

> The 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.[56]

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,[57] 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

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[56] NRC 2003
[57] NRC 2015a

systems.[58] 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.

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 <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 [has the licensee] 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 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

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[58] NRC 2005a

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.

The Panel's conclusions regarding capability of the pressurizer valves to reclose (i.e., avoiding an escalation from an ANS Condition II event to a more serious ANS Condition III event) are applicable to the treatment in the Backfit SE of all three events identified by the NRC staff: the IOECCS event, the CVCS malfunction, and the IOPORV event. For each event, the NRC staff presented scenarios in which pressurizer overfill could result in the PSVs opening, discharging water, and sticking open, which was considered an ANS Condition III event in the Backfit SE. The Panel concludes that the assumption that the PSVs stick open is not necessary for the case of Byron and Braidwood, based on the well-informed technical judgment of prior NRC approvals.

The Panel also observes that Section 15.0.1.2 and Section 15.6 of the Byron and Braidwood UFSAR addresses IOPORV as an ANS Condition II event. The licensee analyzed inadvertent opening of a PSV rather than a PORV, given the PSV's larger relief capacity, and determined that the acceptance criteria for ANS Condition II events had been met. The licensee separately analyzed small break LOCAs in UFSAR Section 15.6.5. The Panel determined that evaluating the appropriateness of the long-standing categorization of this inadvertent-opening event was outside the scope of its review.

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 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 basis 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.[59]

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

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

accordance with the OM Code, or required alternatives to the ASME BPV Code or OM Code, 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 NRC staff's prior approvals reviewed by the Panel, 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'S 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) could 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?

For the specific technical issue reviewed by the Panel (i.e., the potential for pressurizer valves to stick open following water discharge and related issues raised in the Backfit SE), the Panel concluded that the current licensing basis for Byron and Braidwood complies with the applicable regulations and provides adequate protection of the 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's question.

The RES study[60] suggests that the most significant IOECCS sequence, assuming that <u>all</u> pressurizer overfill events would 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 backfit were perfectly effective in preventing a stuck-open PSV.[61] If the PSVs were not assumed to always fail following water discharge (consistent with the NRC staff expert judgment in 2001) or if the backfit were less than perfectly effective, 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

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[60] NRC 2016d

[61] The RES study explains that "any practical backfit remedy is not expected to be completely effective. Therefore, this delta CDF represents the maximum possible benefit from any backfit plant change."

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 discharging 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[62] 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 LAR 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 based on omissions or mistakes of fact.

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[62] ComEd 2000b, Exelon 2001

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 Code 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 draft 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.[63] Because of perceived deficiencies in the rule, the U.S. Nuclear Regulatory Commission (NRC) substantially revised it in 1985.[64] 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.[65] 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.[66] 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.[67]

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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[63] AEC 1970 (Author and year citations in footnotes refer to the designation of references in Appendix D to this report.)
[64] NRC 1985
[65] *Union of Concerned Scientists v. U.S. Nuclear Regulatory Com'n*, 824 F.2d 108, 119-20 (1987).
[66] NRC 1988b
[67] *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:[68]

> 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."[69] 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."[70]

Backfitting Guidance

Extensive information regarding the appropriate implementation of backfitting is provided in NUREG-1409.[71] 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

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

A similar list of examples is provided in Manual Chapter 0514,[72] which is also included as Appendix D to NUREG-1409. Manual Chapter 0514 was referenced in the 1988 rulemaking, and a working draft was provided to the Commission for information in SECY-88-102.[73] 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[74] 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.[75]

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

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[72] NRC 1988c
[73] NRC 1988a
[74] 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.
[75] Imposition of a staff position from which a licensee has previously been excepted is a backfit.

> 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 [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[76] 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.

[76] 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.[77] For example, the ASME BPV Code (2007 Edition) in Article NB-7000, Overpressure Protection, specifies requirements for 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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[77] 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.[78] In March 2007, the NRC staff issued Revision 2 to these SRP sections with significantly more detail, including a statement indicating that PSVs and PORVs can be assumed to reseat properly after discharging water, but only if they have been qualified for water relief.[79]

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.[80] 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.[81] 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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[78] NRC 1981b and NRC 1981c
[79] NRC 2007b and NRC 2007c
[80] NRC 1979a
[81] NRC 1980b and NRC 1980c

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-[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.[82] This report has been referenced in the safety analyses of certain facilities (e.g., North Anna, Units 1 and 2[83]).

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.[84] 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.[85] 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.[86] 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.

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[82] EPRI 1982a
[83] VEPCO 2009, Section 15.2
[84] EPRI 1982b
[85] EPRI 1982c
[86] WOG 1982

In January 1988, Westinghouse issued WCAP-11677, which compared the EPRI test data with feedwater line break safety analyses.[87] 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

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.[88] 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[89] 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.[90] 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.[91]

In Supplement 1 to the Braidwood SER,[92] 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

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[87] Westinghouse 1988
[88] NRC 1983 and NRC 1986b (Braidwood), NRC 1984 and NRC 1987a (Byron)
[89] NRC 1980a
[90] Consumers 1982
[91] ComEd 1982
[92] 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.

similar technical evaluation reports (TERs) developed by Idaho National Engineering Laboratory (INEL).[93] 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 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.

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[93] NRC 1988c (Byron) and NRC 1990a (Braidwood)

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 and its Supplement 1 to operating nuclear power plants (including Byron and Braidwood).[94] 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)[95] 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.[96] 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.[97] In RIS 2005-29, the NRC staff stated that typically Condition II scenarios[98] 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

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[94] Westinghouse 1993 and Westinghouse 1994

[95] Although 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 *Boiler and Pressure Vessel Code*, and by the licensee.

[96] NRC 2003

[97] NRC 2005b

[98] As defined in American Nuclear Society (ANS) Standard 51.1/N18.2-1973 (ANS 1973).

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 to use the PORV block valves to isolate the PORVs is inconsistent with the non-escalation position.

In draft Revision 1 to RIS 2005-29, the NRC staff addresses the specific American Nuclear Society (ANS) Condition II scenarios of chemical volume and control system (CVCS) malfunction, inadvertent opening of a PORV or PSV (IOPSRV), and the IOECCS event.[99] 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 a 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.[100] 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.[101] 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

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[99] NRC 2015a
[100] EPRI 2004
[101] NRC 2011

per minute, respectively. The report summarized EPIX test data for PSVs as no failures to reclose during 1805 demands.

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.[102] 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.[103] 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.[104] 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).[105]

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.[106] 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.

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[102] PG&E 1996
[103] NRC 2004a
[104] PG&E 2003
[105] NRC 1986a
[106] NRC 1997

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 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.[107] 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 to reclose following water discharge.[108]

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.[109] 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.[110] 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.[111] The changes also credited automatic

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[107] NRC 1998
[108] Northeast 1997
[109] EPRI 1982a (Volume 11)
[110] NRC 1989
[111] NRC 2000

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,[112] 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 Callaway's response to TMI Action Plan Item II.D.1[113] 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.[114] 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.[115] 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.[116] The licensee reevaluated its approach and withdrew its TS amendment request.[117] No further action regarding this amendment request was identified by the Panel. However, in a public meeting[118] during the review of the NRR Appeal,[119] 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.

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 to as a stretch power uprate).[120] During its review, the NRC staff requested that the licensee address water solid conditions in the pressurizer because the NRC staff had generally not accepted a solid pressurizer for an IOECCS event 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.[121] The licensee discussed the EPRI testing program in response to TMI Action Plan Item II.D.1, with the results summarized in EPRI NP-2628-SR.[122] The licensee referenced previous NRC approvals related to TMI Action Plan Item II.D.1.[123]

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[112] Union Electric 2000
[113] NRC 1987b
[114] ComEd 1998
[115] NRC 1990b
[116] NRC 1999
[117] ComEd 1999
[118] NRC 2016a
[119] Exelon 2015
[120] NRC 2001a
[121] ComEd 2000b
[122] EPRI 1982b
[123] NRC 1998c and NRC 1990a

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.[124] 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,[125] 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 the 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.

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.[126] The current UFSAR Revision 15 concludes that the IOECCS event does not progress into a stuck-open PSV LOCA event.[127] 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.[128] 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 alternating current 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

[124] Exelon 2001
[125] EPRI 1982c and EPRI 1983
[126] Exelon 2002
[127] Exelon 2014
[128] NRC 2004b

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 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.[129] 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[130] references the ASME *Code for Operation and Maintenance of Nuclear Power Plants* (OM Code), 2004 Edition through 2006 Addenda; and the Braidwood IST program[131] 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).

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[129] NRC 2014a
[130] Exelon 2016b
[131] Exelon 2009

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

- **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).[132] 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)[133] to operate in a water environment were within the valve operable ranges, and thus ensured that the PORVs and SRVs would be operable during the transient. The valve operable ranges were previously determined by the licensee to support operability of the PORVs and SRVs 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.[134] 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

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132 NRC 2001c
133 This term is used in the Shearon Harris SE. The licensee's RAI response (CP&L 2000) makes clear that the referenced SRVs and PORVs are pressurizer valves.
134 NRC 2006

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 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.[135]

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.

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[135] 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 was 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 analyses 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.[136] 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

[136] VEPCO 2009

> 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 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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APPENDIX E: LIST OF ABBREVIATIONS

AEC	Atomic Energy Commission
ALWR	advanced light-water reactor
ANS	American Nuclear Society
ANSI	American National Standards Institute
AOO	anticipated operational occurrence
AP1000	Advanced Passive 1000
ASME	American Society of Mechanical Engineers
BPV Code	*Boiler and Pressure Vessel Code*
BVPS	Beaver Valley Power Station
BWR	boiling-water reactor
CDF	core damage frequency
CFR	*Code of Federal Regulations*
ComEd	Commonwealth Edison
CP&L	Carolina Power & Light Company
CRGR	Committee to Review Generic Requirements
CVCS	chemical and volume control system
ECCS	emergency core cooling system
EDO	Executive Director for Operations
EPIX	Equipment Performance and Information Exchange
EPRI	Electric Power Research Institute
EPU	extended power uprate
ESBWR	Economic Simplified Boiling-Water Reactor
FR	Federal Register
FSAR	final safety analysis report
GDC	General Design Criterion
GL	Generic Letter
HPI	high pressure injection
IAEA	International Atomic Energy Agency
INEL	Idaho National Engineering Laboratory
IOECCS	inadvertent operation of the emergency core cooling system
IOPORV	inadvertent opening of a pressurizer safety or relief valve
IST	inservice testing
JOG	Joint Owners Group
LAR	license amendment request
LLC	limited liability company
LOAC	loss of alternating current power
LOCA	loss-of-coolant accident
MD	Management Directive
MUR	measurement uncertainty recapture
MWt	megawatts thermal
NEI	Nuclear Energy Institute
NRC	Nuclear Regulatory Commission
NRR	Office of Nuclear Reactor Regulation

NSAL	Nuclear Safety Advisory Letter
OM Code	*Code for Operation and Maintenance of Nuclear Power Plants*
PG&E	Pacific Gas and Electric Company
PORV	power-operated relief valve
PSRV	pressurizer safety relief valve
PSV	pressurizer safety valve
PWR	pressurized-water reactor
RAI	request for additional information
RCP	reactor coolant pump
RCS	reactor coolant system
RES	Office of Nuclear Regulatory Research
RIS	Regulatory Issue Summary
RS	Review Standard
SE	safety evaluation
SER	safety evaluation report
SI	safety injection
SOC	statements of consideration
SRP	Standard Review Plan
SRV	safety relief valve
SSC	structure, system, or component
TAC	technical activity code
TER	technical evaluation report
TMI	Three Mile Island
TS	technical specifications
TVA	Tennessee Valley Authority
UFSAR	updated final safety analysis report
VEPCO	Virginia Electric and Power Company
WOG	Westinghouse Owners Group