From: Clark, Theresa
Sent: Wednesday, September 14, 2016 9:54 PM
To: McCree, Victor; Holahan, Gary; Johnson, Michael; Tracy, Glenn
Subject: updated backfit documents
Attachments: NRR memo 9-14-16.docx; NRR memo 9-14-16 tracked.pdf; EDO EXELON BACKFIT APPEAL DECISION - Comm Message Map rev 3 2016 09 14.docx; EDO update.docx

Vic,

Per our discussion this evening:

- I updated the memo to NRR to reflect your changes and a few other minor edits to align the memo with the updated enclosure. Attached are clean and tracked copies, and it is in ADAMS as well. (Note tracked copy is in PDF since tracking doesn't show in Word in MaaS.) I will print in the morning and bring you back all three packages, and can incorporate any additional changes tomorrow.
- I updated the message map with revised dates/steps. Please see attached.
- I drafted a potential EDO Update that incorporates the messages we discussed today, others from the message map, and some additional background on the decision. Please see attached.
- I sent an appointment from my calendar to you, Tim McGinty, John Lubinski, Eric Oesterle, and Jen Whitman for 8:00-8:55 tomorrow. As we discussed, other NRR staff may want to attend as well.

Please let me know how else I can be of assistance. Thanks!

--
Theresa Valentine Clark
Executive Technical Assistant (Reactors)
U.S. Nuclear Regulatory Commission
Theresa.Clark@nrc.gov | 301-415-4048 | O-16E22

MEMORANDUM TO: William M. Dean, Director
Office of Nuclear Reactor Regulation

FROM: Victor M. McCree
Executive Director for Operations

SUBJECT: RESULT OF APPEAL TO THE EXECUTIVE DIRECTOR FOR OPERATIONS OF BACKFIT IMPOSED ON BYRON AND BRAIDWOOD STATIONS REGARDING COMPLIANCE WITH 10 CFR 50.34(b), GDC 15, GDC 21, GDC 29, AND THE LICENSING BASIS

As you are aware, on June 22, 2016, I established a Backfit Appeal Review Panel (Panel) in accordance with Management Directive (MD) 8.4, "Management of Facility-specific Backfitting and Information Collection," to review the subject appeal and to provide me with recommendations (Agencywide Documents Access and Management System (ADAMS) Accession No. ML16173A311). On August 24, 2016, the Panel transmitted the results of its review to me (ADAMS Accession No. ML16236A202). The memorandum from the Panel responding to my tasking, recommended that the 2015 compliance backfit be withdrawn, and included the Panel's report and the basis for this recommendation (ADAMS Accession No. ML16236A208).

I have reviewed the Panel's report, its recommendations, and its responses to the questions I posed when establishing the Panel. In addition, I met with you on September 12, 2016, to discuss my decision and assure that it reflects the thorough, technically sound, and legally well-founded consideration that this matter merits. Our discussion included my response to the additional perspectives you provided to me in your email dated September 2, 2016, which is enclosed, for reference.

As we discussed, the central question in the Panel's review was whether an adequate basis exists for backfitting using the compliance exception in Title 10 of the *Code of Federal Regulations* (10 CFR), Section 50.109(a)4(i) to address potential pressurizer safety valve failures following water discharge. 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...." In answering this question, the Panel focused on the following three related technical and regulatory positions for the pressurizer safety valves (PSVs) described in the staff's October 9, 2015, safety evaluation imposing the backfit (ADAMS Accession No. ML14225A871, referred to as the Backfit SE), as well as the staff's May 3, 2016, response (ADAMS Accession No. ML16095A204) to the backfit appeal by Exelon Generation Company, LLC (the licensee):

1. American Society of Mechanical Engineers (ASME) *Boiler and Pressure Vessel Code* (BPV Code) water qualification (certification) documentation is required if a PSV is to be assumed to reclose after passing water.

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

3. PSVs are subject to the single-failure criterion.

As the Panel noted in its report, it is important to acknowledge that the PSVs in question were designed for steam service and that water discharge through such valves is undesirable and should be minimized or avoided as a matter of conservative engineering and prudent operations. This perspective is reinforced by several industry positions and testing, as well as operator training and control room procedures intended to terminate a potential pressurizer overfill event before filling the pressurizer. For these reasons, the staff's position described in the NRC's backfit imposition letter and its response to the backfit appeal, represents a well-intentioned and conservative approach that could provide additional safety margin. However, based on my review of the relevant documents and discussions, I agree with the Panel's conclusions and support its recommendations. In particular, I agree with the Panel's assessment of the three relevant technical and regulatory positions.

First, regarding ASME Code water qualification (or certification), when considered in the context of the Byron and Braidwood licensing basis, valve "qualification" implies a general demonstration of capability, such as through the Electric Power Research Institute testing conducted in response to Three Mile Island (TMI) Action Plan Item II.D.1, not ASME BPV Code certification. Thus, when preparing the safety evaluations associated with two license amendments in 2001 and 2004 (referred to as 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 in the fully open position. The NRC staff's determination that ASME BPV Code certification is necessary for PSVs first appears in the Backfit SE and is not addressed in any of the final NRC requirements or guidance documents reviewed by the Panel. As such, the NRC staff's position on valve qualification in the Backfit SE represents a new or modified interpretation of what constitutes compliance in addressing potential PSV failures following water discharge.

Second, regarding PSV failure following water discharge, the staff's position in 2001 and 2004, consistent with multiple NRC approvals both before and since, was 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. Without the presumption of PSV failure to reseat, the concerns in the Backfit SE related to event classification, event escalation, and compliance with 10 CFR 50.34(b) and General Design Criteria 15, 21, and 29 are no longer at issue.

Third, the determination that application of the single failure criterion is necessary first appears in the draft Revision 1 to Regulatory Issue Summary 2005-29. This position, which is still under development, is not included in any final NRC requirement or guidance document reviewed by the Panel.

In sum, none of the three positions were "known and established standards of the Commission" when the NRC issued the Uprate SE and the Setpoint SE in license amendments for Byron and Braidwood in 2001 and 2004, respectively, for determining when it was appropriate to assume a failure of a PSV to reseat. Based on the Panel's review, they were not "known and established standards of the Commission" in 2005 (when RIS 2005-29 was issued), in 2006 (when the Beaver Valley extended power uprate was approved), in 2007 (when Revision 2 to Standard Review Plan Sections 15.5.1 – 15.5.2 was issued), nor are they "known and established standards of the Commission" at present.

As a result, I do not support the use of the compliance exception to impose the subject backfit. I agree with the Panel's assessment that the current licensing basis for Byron and Braidwood complies with the applicable regulations and provides adequate protection of public health and safety. I have responded directly to the licensee with my decision on its appeal.

The Panel's report also identifies two issues that warrant further NRC consideration. The report reveals the need to assess the treatment of the underlying technical issue described in the 1993 Westinghouse Nuclear Safety Advisory Letter (NSAL-93-013) on PSV performance after water discharge at pressurized-water reactors. In addition, given the decision communicated herein, the positions included in RIS 2005-29, as well as its proposed Revision 1, should be (re)assessed through the appropriate generic process to ensure they receive appropriate backfit consideration. You are requested to inform me within 120 days of your plan to respond to these issues.

As you are also aware, I have recently directed the U.S. Nuclear Regulatory Commission (NRC) Committee to Review Generic Requirements (CRGR) to assess the adequacy and currency of existing NRC requirements, guidance, criteria, procedures, and training on the subject of backfitting (ADAMS Accession No. ML16133A575). The Panel members have already been in contact with the CRGR to share insights and perspectives from this review. I believe that the CRGR evaluation of our implementation of the backfit process presents us with a timely opportunity to further enhance our regulatory process.

Finally, I recognize that the technical and regulatory positions used in the staff's decision-making involved careful, thorough, and technically solid considerations, reflecting their commitment to ensuring safety. Knowing that our people take seriously the responsibility for assuring public health and safety and are willing to pursue backfits, when appropriate, to assure or enhance safety is key to successfully fulfilling our mission. Although expected, I also sincerely appreciate the cooperation and respect evidenced by both your staff and the Panel members as the Panel evaluated the merits of the licensee's appeal of this technically complex and difficult regulatory issue. Their open, constructive, and collegial interactions reflected the

best of our agency values and contributed to what I consider to be a sound final regulatory decision.

Enclosure:
As stated

cc: Chairman Burns
Commissioner Svinicki
Commissioner Baran

best of our agency values and contributed to what I consider to be a sound final regulatory decision.

Enclosure:
As stated

cc: Chairman Burns
Commissioner Svinicki
Commissioner Baran

DISTRIBUTION: OEDO-16-00584

RidsNrrMailCenter	RidsNrrDorlLpl3-2	AGody, Region II
RidsOgcMailCenter	RidsNrrPMByron	AGendelman, OGC
RidsResMailCenter	RidsNrrPMBraidwood	GMizuno, OGC
RidsNroMailCenter	RidsNrrDss	RCorreia, RES
RidsNmssMailCenter	RidsNrrDe	KSWest, NSIR
RidsRgn1MailCenter	RidsNrrDpr	MBailey, NSIR
RidsRgn2MailCenter	RidsNrrDorl	TScarbrough, NRO
RidsRgn3MailCenter	AGarmoe, NRR	MASpencer, OGC
RidsRgn4MailCenter	TKeene, NRR	TClark, OEDO

ADAMS Accession No.: ML16246A247 EDO-002

OFFICE	OEDO	OEDO	OEDO
NAME	GHolahan	MJohnson	VMcCree
DATE	09/12/16	09/12/16	09/ /16

OFFICAL RECORD COPY

RESPONSE BY EXECUTIVE DIRECTOR FOR OPERATIONS (EDO) TO ADDITIONAL PERSPECTIVES ON BACKFIT APPEAL REVIEW PANEL FINDINGS PROVIDED IN SEPTEMBER 2, 2016, EMAIL FROM THE DIRECTOR OF THE OFFICE OF NUCLEAR REACTOR REGULATION (NRR)

1. **NRR appreciates the panel's efforts. However, NRR believes that the panel's perspectives do not provide sufficient basis to overturn the backfit.**

   Response: Based on my review, the Panel's perspectives provide a sound basis for supporting the licensee's appeal of the compliance exception backfit. The concerns listed below do not address the specific Panel finding that is a primary basis for overturning the backfit. In particular, the NRC has previously accepted water qualification of pressurizer safety valves (PSVs) and power-operated relief valves (PORVs) based on Electric Power Research Institute (EPRI), Wyle, or vendor testing for nuclear power plants (beyond Byron and Braidwood) as part of Three Mile Island (TMI) action items, Chapter 15 accident analyses, and other evaluations. In those evaluations, the NRC did not require American Society of Mechanical Engineers (ASME) *Boiler and Pressure Vessel Code* (BPV Code) certification for water service.

2. **NRR Concerns**

   a. **The panel has narrowly focused its review on the water qualification question. NRR maintains that the original backfit documents numerous issues with the licensing basis for Byron and Braidwood that have not been addressed in the panel's assessment.**

      Response: In the report, I find that the Panel adequately addressed the issues identified as important by NRR in its comments on the preliminary findings. Although NRR raised several issues of concern, I find that the most salient positions and issues associated with the compliance exception backfit question have been appropriately addressed.

   b. **With regard to the PSV water qualification question, the panel's position is reliant on its interpretation of the 1977 Information SECY [SECY-77-439, "Single Failure Criterion," dated August 17, 1977]. The panel has provided select quotes from that SECY that it believes supports its position. NRR believes that when the entire SECY is reviewed it becomes clear that the SECY was simply documenting current practices in 1977, some of which were still being researched, and does not provide a "known and established standard." The staff contends that if the 1977 SECY had been intended to provide the "known and established standard" it would have been included in subsequent updates to regulations, regulatory guides, and SRPs [Standard Review Plans] over the following nearly 40 years. It has not.**

      Response: In its report, the Panel indicates that it addressed SECY-77-439 in response to NRR's assertion that Exelon had not satisfied the "single failure assumption" for the PSVs at Byron and Braidwood. I find, based on the Panel's evaluation, that there is not sufficient definition regarding the "known and established standards of the Commission" to justify use of the "single failure assumption" to support the compliance backfit exception.

c. **In numerous places the panel quotes documents that it interprets as describing the treatment of check valves as analogous to PSVs. The panel did not find any definitive documentation that demonstrates that the agency concluded that PSVs are analogous to check valves and, as such, should be considered passive components. This appears to be the panel's judgement, not an NRC position. NRR disagrees with the panel's interpretation and has historically treated PSVs as active components, including designating them as such during license renewal. PSVs are designed to perform a specific [reactor coolant system] overpressure protection safety function critical to protecting one of the key defense-in-depth barriers to protect public health and safety from the release of radioactive materials. The staff believes the panel's comparison is inappropriate and establishes a very concerning precedent.**

   Response: The Panel's discussion of check valves in relation to PSVs was provided for context given that PSVs were not explicitly discussed in documents describing passive failures and the application of the single failure criterion (e.g., SECY-77-439; SECY-94-084, "Policy and Technical Issues Associated with the Regulatory Treatment of Non-Safety Systems in Passive Plant Designs," dated March 28, 1994; and SECY-05-0138, "Risk-Informed and Performance-Based Alternatives to the Single-Failure Criterion," dated August 2, 2005). The Panel's discussion highlights that there is no definitive documentation on how PSVs should be treated and that the panel did not intend to establish precedent for treatment of PSVs by providing this context. I find, based on the Panel's evaluation, that there is not sufficient definition regarding the "known and established standards of the Commission" associated with the treatment of these valves as passive or active. As such, the threshold for meeting the compliance backfit exception was not met.

d. **On page 13, the panel acknowledges the Byron/Braidwood licensing basis as categorizing the PSVs and PORVs as active components. However, the panel, given its reliance on treating PSVs akin to check valves, establishes a new and different position in its own summary when it determines these valves should be treated as passive components for the purposes of considering the single failure criterion.**

   Response: Page 13 of the Panel's report discusses several different possibilities for evaluating the PSVs and quotes SECY-94-084. As discussed in the response to item 2c, the Panel did not intend to establish precedent for treatment of PSVs. However, I find, based on the Panel's evaluation, that there is not sufficient definition regarding the "known and established standards of the Commission" associated with the treatment of these valves as passive or active. As such, the threshold for meeting the compliance backfit exception was not met.

e. **Regarding ASME, [Title 10 of the *Code of Federal Regulations* (10 CFR), Section] 50.55a requires nuclear power plants to be initially designed and constructed [in accordance with] ASME [BPV Code], Section III and to be tested throughout their service life [in accordance with] ASME OM Code [*Operation and Maintenance of Nuclear Power Plants*]. These codes comprise the qualification standards for ASME Class 1 safety valves such as the pressurizer PSVs with which licensees**

are required to comply unless alternatives have been authorized by the staff [in accordance with] 10 CFR 50.55a.

Response: The ASME BPV Code does indeed provide certification requirements for safety and relief valves for their intended design function. However, as noted in the Panel's report, there are a number of examples where the NRC has accepted qualification of safety or relief valves based on EPRI, Wyle, and vendor testing. The staff's position in the backfit is different from these past approvals. Based on the absence of a "known and established standard of the Commission" requiring ASME BPV Code certification for the water discharge conditions that are the subject of the backfit, I find that the threshold for meeting the compliance backfit exception was not met.

f. **The panel asserts in its summary that the valves in question were water qualified due to the licensee's reliance on them to pass water during feedline break events. The panel does not appear to acknowledge that feedline breaks are Condition IV events, similar to [loss-of-coolant accidents], which are never expected to occur in the lifetime of the facilities and therefore, given their lower probability of occurrence, are permitted to have more significant consequences. The EPRI testing demonstrated acceptable performance under conditions anticipated during these Condition IV events (higher temperature fluid ~ 650°F), while the EPRI test at the more likely Condition II inadvertent mass addition event conditions (lower temperature fluid ~550°F) was terminated early due to valve chatter on opening. The summary of the EPRI testing indicated that for subcooled water conditions valve chatter and resultant valve damage was generally observed.**

Response: This comment suggests that the EPRI testing did not address water discharge for the Byron and Braidwood PSVs at an acceptable water temperature. Although the issue raised by NRR regarding the adequacy of the EPRI testing may be considered in a future generic review, it is not pertinent to the central issue concerning the existence of a "known and established standard."

3. **Path Forward**

a. **If the EDO supports the original backfit, NRR agrees with the panel that risk insights are important considerations in determining how reasonable assurance of compliance can be demonstrated. However, as acknowledged by the panel, consistent with [Regulatory Guide (RG)] 1.174, risk insights must include consideration of defense-in-depth and safety margins. If a PSV were to stick open or significantly leak at Bryon and Braidwood during a licensing basis Condition II event, which is anticipated to occur on an annual frequency, the licensee has not yet demonstrated adequate defense-in-depth. NRR is open to considering risk-informed licensing basis changes, or potential plant modifications, that appropriately consider all 5 elements of RG 1.174.**

Response: I support the recommendations of the Panel. However, I agree that the safety significance of the potential for the PSVs to stick open should be considered as part of a generic resolution of this issue for all pressurized-water reactors.

b. **If the EDO supports the Backfit Panel's conclusion, NRR requests that the EDO allow the staff to independently assess what path forward is appropriate given the positions documented in the panel's report and EDO's decision. In particular,**

NRR has concerns regarding the recommendations on page 3 of the report that need to be further considered before determining what future course of action is most appropriate.

Response: I agree. The report reveals the need to assess the treatment of the underlying technical issue described in the 1993 Westinghouse Nuclear Safety Advisory Letter (NSAL-93-013) on PSV performance after water discharge at pressurized-water reactors. In addition, given the decision communicated herein, the positions included in Regulatory Issue Summary 2005-29, as well as its proposed Revision 1, should be (re)assessed through the appropriate generic process to ensure they receive appropriate backfit consideration. The Director of NRR should inform me within 120 days of the plan to respond to these issues.

MEMORANDUM TO: William M. Dean, Director
Office of Nuclear Reactor Regulation

FROM: Victor M. McCree
Executive Director for Operations

SUBJECT: RESULT OF APPEAL TO THE EXECUTIVE DIRECTOR FOR OPERATIONS OF BACKFIT IMPOSED ON BYRON AND BRAIDWOOD STATIONS REGARDING COMPLIANCE WITH 10 CFR 50.34(b), GDC 15, GDC 21, GDC 29, AND THE LICENSING BASIS

As you are aware, on June 22, 2016, I established a Backfit Appeal Review Panel (Panel) in accordance with Management Directive (MD) 8.4, "Management of Facility-specific Backfitting and Information Collection," to review the subject appeal and to provide me with recommendations (Agencywide Documents Access and Management System (ADAMS) Accession No. ML16173A311). On August 24, 2016, the Panel transmitted the results of its review to me (ADAMS Accession No. ML16236A202). The memorandum from the Panel responding to my tasking, recommended that the 2015 compliance backfit be withdrawn, and included the Panel's report and the basis for this recommendation (ADAMS Accession No. ML16236A208).

I have reviewed the Panel's report, its recommendations, and its responses to the questions I posed when establishing the P~~p~~anel. In addition, I met with you on September 12, 2016, to discuss my decision and assure that it reflects the thorough, technically sound, and legally well-founded consideration that this matter merits. Our discussion included my response to the additional perspectives you provided to me in your email dated September 2, 2016, which is enclosed, for reference.

As we discussed, the central question in the Panel's review was whether an adequate basis exists for backfitting using the compliance exception in Title 10 of the *Code of Federal Regulations* (10 CFR), Section 50.109(a)4(i) to address potential pressurizer safety valve failures following water discharge. 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...." In answering this question, the Panel focused on the following three related technical and regulatory positions for the pressurizer safety valves (PSVs) described in the staff's October 9, 2015, safety evaluation imposing the backfit (ADAMS Accession No. ML14225A871, referred to as the Backfit SE), as well as the staff's May 3, 2016, response (ADAMS Accession No. ML16095A204) to the backfit appeal by Exelon Generation Company, LLC (the licensee):

1. American Society of Mechanical Engineers (ASME) *Boiler and Pressure Vessel Code* (BPV Code) water qualification (certification) documentation is required if a PSV is to be assumed to reclose after passing water.

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

3. PSVs are subject to the single-failure criterion.

As the Panel noted in its report, it is important to acknowledge that the PSVs in question were designed for steam service and that water discharge through such valves is undesirable and should be minimized or avoided as a matter of conservative engineering and prudent operations. This perspective is reinforced by several industry positions and testing, as well as operator training and control room procedures intended to terminate a potential pressurizer overfill event before filling the pressurizer. For these reasons, the staff's position described in the NRC's backfit imposition letter and its response to the backfit appeal, represents a well-intentioned and conservative approach that could provide additional safety margin. However, based on my review of the relevant documents and discussions, I agree with the Panel's conclusions and support its recommendations. In particular, I agree with the Panel's assessment of the three relevant technical and regulatory positions.

First, regarding ASME Code water qualification (or certification), when considered in the context of the Byron and Braidwood licensing basis, valve "qualification" implies a general demonstration of capability, such as through the Electric Power Research Institute testing conducted in response to Three Mile Island (TMI) Action Plan Item II.D.1, not ASME BPV Code certification. Thus, when preparing the safety evaluations associated with two license amendments in 2001 and 2004 (referred to as 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 in the fully open position. The NRC staff's determination that ASME BPV Code certification is necessary for PSVs first appears in the Backfit SE and is not addressed in any of the final NRC requirements or guidance documents reviewed by the Panel. As such, the NRC staff's position on valve qualification in the Backfit SE represents a new or modified interpretation of what constitutes compliance in addressing potential PSV failures following water discharge.

Second, regarding PSV failure following water discharge, the ~~standard in place~~staff's position in 2001 and 2004, consistent with multiple NRC approvals both before and ~~at present, is~~since, was 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. Without the presumption of PSV failure to reseat, the concerns in the Backfit SE related to event classification, event escalation, and compliance with 10 CFR 50.34(b) and General Design Criteria 15, 21, and 29 are no longer at issue.

Third, the determination that application of the single failure criterion is necessary first appears in the draft Revision 1 to Regulatory Issue Summary 2005-29. This position, which is still under development, is not included in any final NRC requirement or guidance document reviewed by the P~~p~~anel.

In sum, ~~neither~~none of the three positions were "known and established standards of the Commission" when the NRC issued the Uprate SE and the Setpoint SE in license amendments for Byron and Braidwood in 2001 and 2004, respectively, for determining when it was appropriate to assume a failure of a PSV to reseat. Based on the Panel's review, they were not "known and established standards of the Commission" in 2005 (when RIS 2005-29 was issued), in 2006 (when the Beaver Valley extended power uprate was approved), in 2007 (when Revision 2 to Standard Review Plan Sections 15.5.1 – 15.5.2 was issued), nor are they "known and established standards of the Commission" at present.

As a result, I do not support the use of the compliance exception to impose the subject backfit. I agree with the Panel's assessment that the current licensing basis for Byron and Braidwood complies with the applicable regulations and provides adequate protection of public health and safety. I have responded directly to the licensee with my decision on its appeal.

The Panel's report also identifies two issues that warrant further NRC consideration. The report reveals the need to assess the treatment of the underlying technical issue described in the 1993 Westinghouse Nuclear Safety Advisory Letter (NSAL-93-013) on PSV performance after water discharge at pressurized~~-~~-water reactors. In addition, given the decision communicated herein, the positions included in RIS 2005-29, as well as its proposed Revision 1, should be (re)assessed through the appropriate generic process to ensure they receive appropriate backfit consideration. You are requested to inform me within ~~30~~120 days of your plan to respond to these issues.

As you are also aware, I have recently directed the U.S. Nuclear Regulatory Commission (NRC) Committee to Review Generic Requirements (CRGR) to assess the adequacy and currency of existing NRC requirements, guidance, criteria, procedures, and training on the subject of backfitting (ADAMS Accession No. ML16133A575). The Panel members have already been in contact with the CRGR to share insights and perspectives from this review. I believe that the CRGR evaluation of our implementation of the backfit process presents us with a timely opportunity to further enhance our regulatory process.

Finally, I recognize that the technical and regulatory positions used in the staff's decision-making involved careful, thorough, and technically solid considerations, reflecting their commitment to ensuring safety. Knowing that our people take seriously the responsibility for assuring public health and safety and are willing to pursue backfits, when appropriate, to assure or enhance safety is key to successfully fulfilling our mission. Although expected, I also sincerely appreciate the cooperation and respect evidenced by both your staff and the Panel members as the Panel evaluated the merits of the licensee's appeal of this technically complex and difficult regulatory issue. Their open, constructive, and collegial interactions reflected the

best of our agency values and contributed to what I consider to be a sound final regulatory decision.

Enclosure:
As stated

cc: Chairman Burns
Commissioner Svinicki
Commissioner Baran

best of our agency values and contributed to what I consider to be a sound final regulatory decision.

Enclosure:
As stated

cc: Chairman Burns
Commissioner Svinicki
Commissioner Baran

DISTRIBUTION: OEDO-16-00584

RidsNrrMailCenter	RidsNrrDorlLpl3-2	AGody, Region II
RidsOgcMailCenter	RidsNrrPMByron	AGendelman, OGC
RidsResMailCenter	RidsNrrPMBraidwood	GMizuno, OGC
RidsNroMailCenter	RidsNrrDss	RCorreia, RES
RidsNmssMailCenter	RidsNrrDe	KSWest, NSIR
RidsRgn1MailCenter	RidsNrrDpr	MBailey, NSIR
RidsRgn2MailCenter	RidsNrrDorl	TScarbrough, NRO
RidsRgn3MailCenter	AGarmoe, NRR	MASpencer, OGC
RidsRgn4MailCenter	TKeene, NRR	TClark, OEDO

ADAMS Accession No.: ML16246A247 EDO-002

OFFICE	OEDO	OEDO	OEDO
NAME	GHolahan	MJohnson	VMcCree
DATE	09/—12/16	09/—12/16	09/ /16

OFFICAL RECORD COPY

Page 016 of 340

Withheld pursuant to exemption

(b)(5)

of the Freedom of Information and Privacy Act

Page 017 of 340

Withheld pursuant to exemption

(b)(5)

of the Freedom of Information and Privacy Act

Page 018 of 340

Withheld pursuant to exemption

(b)(5)

of the Freedom of Information and Privacy Act

Page 019 of 340

Withheld pursuant to exemption

(b)(5)

of the Freedom of Information and Privacy Act

(b)(5)

Communications Message Map:	EDO EXELON BACKFIT APPEAL DECISION:	SEPTEMBER 12, 2016
KEY MESSAGES	**NRC ACTIONS/ACTIVITIES**	**BACKGROUND**
The staff takes its responsibility for assuring safety very seriously; and pursues technically sound, and legally well-founded backfits when it concludes they are needed to assure or enhance safety. On complex technical and legal matters there can be differing views either within the staff, or with licensees and other stakeholders. The NRC used its formal backfit review process to ensure this issue received appropriate consideration. In this case, the EDO concluded that the NRC staff's position on valve qualification, valve performance, and the application of the single failure criterion in the backfit safety evaluation was a new or modified interpretation of what constitutes compliance in addressing potential pressurizer safety valve failures following water discharge, and did not provide a basis for a compliance backfit. Previously (on June 9, 2016) the EDO tasked the Committee to Review Generic Requirements (CRGR) to review NRC implementation of agency backfitting and finality guidance. This effort, which includes an assessment of the clarity and effectiveness of backfitting requirements, guidance and criteria, staff training and knowledge management, will incorporate insights from the EDO's decision on this matter.	On June 22, 2016, the EDO established a Backfit Appeal Review Panel (Panel) of senior staff and managers to review the Exelon backfit appeal. On August 24, 2016, the Panel recommended, and the EDO supported, a reversal of the compliance backfit, agreeing with the Exelon appeal. **NEXT STEPS** The EDO will verbally inform the Chairman and Commissioners of this decision. **Date: September 13, 2016 (am)** The EDO will send a memorandum to the Director, NRR, formally notifying him of this decision. **Date: September 15, 2016 (am)** The EDO will call Exelon to inform them of his decision. **Date: September 15, 2016 (pm)** The EDO will send a letter informing the licensee (Exelon) of this decision. **Date: September 15, 2016 (pm)** The EDO will call NEI to inform them of his decision. **Date: September 15, 2016 (pm)** The EDO will send a letter to NEI in response to its earlier letter supporting Exelon's backfit appeal. **Date: September 15, 2016 (pm)** The EDO will issue an EDO Update communicating key points to the staff. **Date: September 16, 2016 (am)** OPA, in coordination with OEDO, will issue a blog post announcing this decision: **September 16, 2016 (pm)**	The NRC staff issued a compliance backfit letter to Exelon (October 9, 2015) on the issue of pressurizer overfill and safety valve performance during Condition II events (ANS Condition II categorization as frequent events). Exelon twice appealed the compliance backfit (once at the office level, then at the EDO level) claiming it was inappropriate since, in their view, the staff failed to identify any error or omission that make the previously approved analysis incorrect. The NRC regulation for backfitting (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...."

Last week, I shared with you some insights about our responsibility to assure adequate protection of public health and safety. This assurance can include the need to impose backfits in accordance with 10 CFR 50.109—as we did, for example, following the accident at Fukushima Dai-ichi.

Backfitting can only be done, however, when certain criteria laid out in the rule are met. Our agency makes careful, thorough, and technically solid regulatory findings when licensing nuclear power plants. As such, we should not change our findings and impose new requirements without an appropriate basis.

Recently, I issued a final decision that supported a licensee appeal and overturned a backfit imposed by the Office of Nuclear Reactor Regulation (NRR). This backfit was issued in 2015 for the Byron and Braidwood nuclear plants in Illinois.

The technical issue in this backfit relates to certain accident scenarios where the pressurizer could overfill. In these cases, water would be discharged from the pressurizer safety valves, which are designed to pass steam instead of liquid water. The staff approved a certain approach to these scenarios in 2001 and 2004, and concerns were later raised with these approvals. The regulatory and legal issue is whether the "compliance exception" to the Backfit Rule (10 CFR 50.109) requirement to conduct a backfit analysis was properly applied. In short, the staff would need to show that the initial approvals had been based on a mistake or omission, not that the interpretation of what was acceptable changed over time.

Following our Management Directive 8.4 on backfits, I chartered a panel of senior technical and legal staff to consider the facts in this case and recommend a response. This panel reviewed over a hundred documents related to the plants' licensing basis and the history of the technical issues in question and provided a detailed report [link to ML16236A208]. Considering both this report and discussions I had with NRR staff who contributed to the backfit, I determined that the positions taken in the backfit were a new or modified interpretations of what constitutes compliance in addressing potential pressurizer safety valve failures following water discharge, and did not provide a basis for a compliance backfit.

Although I decided to support the licensee's appeal in this case, I am proud to know that our people take seriously the responsibility for assuring public health and safety and are willing to pursue backfits, when appropriate. I encourage the staff to continue to raise issues of potential safety significance, adequate protection, and compliance.

Furthermore, I recognize that we need to fully understand and disposition the technical concerns that underlie this backfit for the larger group of licensees to which they apply. Therefore, I have referred these technical issues to NRR [link to ML16246A247] for further assessment and have asked that a plan be provided within 120 days.

From: Scarbrough, Thomas
Sent: Thursday, August 04, 2016 8:52 AM
To: Holahan, Gary; West, Steven; Clark, Theresa; Spencer, Michael
Subject: RE: Things to review
Attachments: Backfit Appeal Panel Report 2016 07 13 - Scarbrough - R2 (Enclosures 2 and 3) 8-4-2016.docx

Gary,

In your e-mail, you indicate that you will begin working on the Enclosures.

Attached is my August 4 version of Enclosures 2 and 3 for the draft report. I also included a copy in the S:/ drive Report folder.

FYI, I included a conclusion regarding the Braidwood/Byron licensee actions at the end of Enclosure 3 based on my previous list of licensee questions.

Thanks.
Tom

From: Holahan, Gary
Sent: Wednesday, August 03, 2016 4:58 PM
To: West, Steven <Steven.West@nrc.gov>; Scarbrough, Thomas <Thomas.Scarbrough@nrc.gov>; Clark, Theresa <Theresa.Clark@nrc.gov>; Spencer, Michael <Michael.Spencer@nrc.gov>
Subject: Things to review

Steve,
Tom,
Michael,
Theresa,

I have taken the Preliminary Findings document and incorporated it into a "Discussion" section. I added text to make it read like a report (no changes to findings or conclusions). Please review at Reports / Backfit Appeal Report 2016 08 03 2pm.

Next I will start on the Enclosures (and the sections referring to them).

I have also drafted (first draft...) a memo to Vic presenting the report. Please review at Reports / Cover memo Backfit Appeal Panel 2016 08 03 2pm.

Gary

P.S. I told Vic that I promoted you to team member, so he won't be surprise to see your name on the cover memo.

T. Scarbrough
August 4, 2016

ENCLOSURE 2

Qualification of Pressure Relief Valves in Nuclear Power Plants in Response to TMI-2 Accident

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

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

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

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

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

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

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

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

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

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

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

Final BWR Test Program--October 1, 1980

Block Valve Qualification Program--January 1, 1981

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

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

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

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

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

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

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

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

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

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

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

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

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

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

In January 1988, WCAP-11677, "Pressurizer Safety Relief Valve for Water Discharge During a Feedwater Line Break," provided a description of the WOG comparison of the EPRI test data with feedline break safety analyses. This report was submitted as an attachment to a response to a request for additional information (RAI) dated May 8, 1989, from the licensee of the Seabrook nuclear power plant. (ADAMS Microfiche 49775:336 – 49756:017) As discussed in the report, the WOG determined that all nuclear power plants addressed in the EPRI testing have PSVs that will operate reliably during water relief. The WOG evaluated the performance of the Crosby 6M6 PSVs during the EPRI tests, and considered that the performance involved less significant flutter (half lift motion) than the chatter (full lift motion) determined in the EPRI report. The WOG concluded that the Crosby 6M6 PSV can pass slightly subcooled water at a minimum up to 3 times without damage.

ENCLOSURE 3

Concerns regarding Performance of Pressurizer Valves under Water Flow Conditions

In 1993 and 1994, Westinghouse issued Nuclear Safety Advisory Letter NSAL-93-013 (June 30, 1993) and NSAL-93-013, Supplement 1 (October 28, 1994) under 10 CFR Part 21 to operating nuclear power plants (including Braidwood and Byron) in response to its discovery that potentially nonconservative assumptions were used in the licensing analysis of the Inadverent Operation of the Emergency Core Cooling System at Power (IOECCS) event. In this NSAL, Westinghouse recommended that licensees determine if their Pressurizer Safety Relief Valves (PSRVs) are capable of closing following discharge of subcooled water. Westinghouse noted that the PSRVs might have been designed or "qualified" to relieve subcooled water. Westinghouse indicated that water relief through the Power-Operated Relief Valves (PORVs) is not a concern, because the PORV block valves can be used to isolate the PORVs if they fail to close. If the PSRVs are not designed or qualified for subcooled water relief, Westinghouse recommended that licensees re-evaluate the IOECCS event with three possible options of (1) reducing ECCS flow used in the safety analysis, (2) using a less restrictive operator response time, or (3) crediting the use of one or more PORVs to help mitigate the event. In Supplement 1 to NSAL-93-013, Westinghouse alerted licensees to a potential reduced time for operator action if a positive displacement pump is in service, and to the need to qualify the PSRVs and the piping downstream of the PSRVs and PORVs if water relief from the pressurizer is predicted.

Some licensees of operating nuclear power plants alerted the NRC to their actions to address the potential concerns regarding liquid service for pressurizer safety valves (PSVs) and PORVs. A sample of actions by nuclear power plant licensees is summarized below:

On August 13, 1996, the licensee of the Diablo Canyon nuclear power plant submitted a report under 10 CFR 50.59 related to the potential for an IOECCS event. (ADAMS Microfiche 89419:294-322) The submittal included NSAL-93-013 and its supplement as enclosures. The licensee indicated that the PSVs had not been initially qualified for water relief, but were subsequently qualified for a brief period. The licensee indicated that WCAP-11677 was applicable and demonstrated that the PSVs were operable. On July 2, 2004, the NRC granted a license amendment request (LAR) for Diablo Canyon that allowed credit for actuation of the PORVs in response to inadvertent safety injection (SI) actuation to avoid challenges to the PSVs. (ADAMS Accession No. ML041950300) In support of that LAR, the licensee responded on November 21, 2003, to requests for additional information (RAIs) related to the capability of the PORVs to function adequately under conditions predicted for design-basis transients and accidents. (ADAMS Accession No. ML033360735) In response to an RAI regarding the design adequacy of the PORVs if the pressurizer becomes water solid, the licensee stated that the NRC had issued a letter dated January 26, 1986, "Safety and Relief Valve Testing, NUREG-0737 Item II.D.1," that provided an SER that accepted the adequacy of the PORV and block valve design and confirmatory testing for a range of fluid conditions (full pressure steam, steam to water transition, and subcooled water fluid).

On June 4, 1997, the NRC granted a technical specification (TS) revision for the Salem nuclear power plant to ensure that the automatic capability of the PORVs to relieve pressure is maintained. (ADAMS Accession No. ML011720397) In response to NSAL 93-013, the licensee determined that an inadvertent SI actuation at power could cause the pressurizer to become water solid and PSVs lifting with water relief if the automatic operation of the PORVs is not made available for reactor coolant system (RCS) depressurization early in the transient. In that the Salem PSVs were not designed to relieve water, it was noted that water relief has the potential to cause the PSVs to fail in the open position. In the course of the review of the licensee's application, the NRC staff noted that the PORVs were not designed to "safety related" standards and, thus, could not be credited for mitigation of the inadvertent SI actuation at power incident when the PORV is operating in the automatic mode. In response, the licensee proposed an upgrade of the PORVs to eliminate the possibility that a single active failure of a PORV component could prevent the mitigation of the inadvertent SI actuation at power incident. As discussed in the SER, the licensee implemented modifications to the PORV circuitry to qualify the upgraded circuitry as safety-related. Regarding PORV performance, the licensee evaluated the PORV air accumulators for sufficient capacity for the inadvertent SI event. The licensee also reported that endurance tests had been performed with five different trims (with different trim materials) on one PORV at Wyle Laboratories to demonstrate that (1) after 2000 consecutive operations, there were no packing leaks nor packing gland adjustments required; (2) there was no diaphragm failure; and (3) the solenoid valve withstood 10,000 operations without any loss of function. Based on this information, the NRC staff concluded that the PORV performance was acceptable regarding the mitigation of an inadvertent SI event.

On June 5, 1998, the NRC granted a license amendment for Millstone Unit 3 for a TS revision to ensure that the capability of the PORVs to relieve pressure is maintained. (ADAMS Accession No. ML011800207) The revised TS Bases stated that the PORVs and their associated piping have been demonstrated to be "qualified" for water relief. The PORVs prevent water relief from the PSVs for which qualification for water relief has not been demonstrated. The TS Bases also stated that the prime importance for the capability to close the block valve is to isolate a stuck-open PORV. In the SER, the staff stated that the licensee notified the NRC of the issue of potential water relief through the PSVs that could lead to valve failure in LER 97-063-00 on December 31, 1997. To provide added assurance that the PSVs will not be damaged due to water relief during an ISI event, the licensee upgraded the PORV circuitry, added additional PORV surveillance requirements, qualified the PORVs and associated piping for water relief, and made emergency procedure changes to allow plant operators additional time to terminate the event. With respect to the PORV circuitry, the staff concluded that the PORV circuitry modifications qualified the PORV control circuitry as safety-related. With respect to PORV performance, the licensee reanalyzed the inadvertent SI event with the LOFTRAN computer code to demonstrate that the PORVs were qualified for water relief for approximately 1 hour. The licensee referenced EPRI testing documented in NP-2670-LD, Volume 11, that was said to generically resolve post TMI-2 issues associated with PORVs and safety valve qualification for water and steam relief, with the results from four tests of a Garrett PORV (such as used at Millstone Unit 3) for water relief. The licensee determined that the PORVs and associated piping are qualified for 1 hour of water relief for inadvertent operation of the ECCS (IOECCS). The licensee also stated that the PORV manufacturer performed numerous cycle tests to verify the performance of the valve design, and also verified that valve seat leakage was acceptable. The licensee stated that the PORV block valves had been evaluated for water relief in accordance with the program established in response to Generic Letter (GL) 89-10, "Safety-Related Motor-Operated Valve Testing and Surveillance." The NRC staff found the licensee information regarding the qualification of the PORVs for water relief during the ISI event to be acceptable.

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

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

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

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

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

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

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

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

In August 2004, EPRI issued Report 1011047, "Probability of Safety Valve Failure-to-Reseat Following Steam and Liquid Relief - Quantitative Expert Elicitation," which evaluated the potential increase in failure rates following steam and liquid relief through safety valves based on expert judgement. The report found that the increase in failure rates is difficult to estimate because of limited data. However, the experts considered that repeated water relief through safety valves might cause increased chatter, and therefore, an increased failure rate.

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

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

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

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

Several licensees have addressed the performance of PSVs and PORVs as part of LARs or in their UFSARs. Some of those PSV and PORV evaluations are summarized below:

On October 12, 2001, the NRC granted a license amendment to the Shearon Harris nuclear power plant for steam generator replacement and a power uprate to a maximum power level of 2900 MWt (approximately 4.5%). In addressing the licensee's evaluation of SRP Section 15.5.1, the staff found that the analysis showed that the calculated inlet pressures and temperatures required for the PORVs and SRVs to operate in a water environment are within the valve operable ranges, and thus ensure that the PORV and SRV are operable during the transient. The valve operable ranges were previously determined by the licensee to support operability of the PORV and SRV during the discharge of subcooled water in accordance with the NUREG-0737 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 DBNR limit, the staff concluded that the analysis was acceptable.

On July 19, 2006, the NRC granted an EPU to Beaver Valley Units 1 and 2 (BVPS-1 and 2) for an approximate 8% increase in thermal power to 2,900 MWt. In its SER (ADAMS No. ML061720376), the staff stated that a specific issue which was reviewed related to the capability of the PSVs to discharge liquid and adequately reseat for a spurious SI actuation. The specific issue which the staff evaluated in this regard was whether the PSVs could reasonably be expected to reseat in order to prevent the spurious SI actuation (a Condition II event) from causing a stuck-open PSV (a Condition III event). This issue was said to be further discussed in RIS 2005-29. While the PSVs are qualified to discharge steam, if the valves discharge liquid having a temperature low enough, they may not reseat properly. Based the licensee's analysis, during a spurious SI event, the PSVs would be required to discharge steam followed by high temperature liquid after the pressurizer fills. The licensee provided plots of the pressurizer water temperatures for this event which indicated that the minimum temperature of the discharged liquid for both BVPS-1 and 2 is approximately 620 °F. To evaluate the capability of the valves to discharge and reseat, the staff reviewed the available data from the full flow tests performed during the EPRI test program in 1981 for the specific PSV models representative of those installed at BVPS-1 and 2. The licensee also used the methodology contained in WCAP-11677, and determined that the minimum acceptable liquid temperature for which the PSVs are expected to successfully discharge and reseat is less than the minimum expected temperature for the spurious SI event for BVPS-1 and 2. The staff agreed that both the minimum expected liquid discharge temperature and the minimum acceptable liquid temperature had been conservatively calculated. Therefore, the staff determined that, for purposes of preventing the occurrence of a more serious Condition III event, there is reasonable assurance that the PSVs would adequately discharge and reseat following a spurious SI actuation. A consideration in making this finding was that, in the unlikely event of a stuck-open PSV, the ECCS is fully capable of mitigating the resulting LOCA.

On June 15, 2012, the NRC granted an EPU for Turkey Point (PTN) Units 3 and 4 that increased the thermal power level of each unit approximately 15% to 2644 MWt. In the SER (ADAMS Accession No. ML11293A359), the staff indicated that ECCS actuation is not a possible initiator of inadvertent increase in reactor coolant inventory because the high head SI pumps have a shut-off head below the normal RCS operating pressure. The staff stated that a CVCS malfunction that increases RCS inventory was evaluated for the effects of adding water inventory to the RCS. If the pressurizer fills and causes water to be relieved through the PORVs or safety valves, then these valves could stick open and create a small break LOCA. The staff stated that this would violate the acceptance criterion that prohibits the escalation of an anticipated operational occurrence (AOO) into a more serious event. Satisfaction of this acceptance criterion is demonstrated by showing that sufficient time exists for the operator to recognize the situation and end the charging flow before the pressurizer can fill. The staff concluded that the licensee's analyses of IOECCS and CVCS events adequately accounted for operation of the plant at the proposed power level. Regarding an inadvertent opening of a pressurizer relief valve, the licensee initially proposed that the consequences of this event are bounded by the small break LOCA. The staff did not accept this proposed disposition. If action is not taken to secure the open valve by either closing the PORV or its block valve, the staff stated that this event could escalate to a small break LOCA, which is contrary to the non-escalation criterion set forth for AOOs. When the pressurizer becomes water solid, water begins to flow through the open PORV. If the PORV is not qualified for water relief, the staff stated that it is likely the PORV will not close upon demand. In this way, the staff stated that the inadvertent opening of a PORV, an AOO, becomes a small break LOCA at the top of the pressurizer, a Condition III event. The staff requested that the licensee address the inadvertent opening of the PORV with respect to the third criterion for a Condition II event. The licensee provided an analysis, performed largely in accordance with NRC-approved, Westinghouse analytic methodology using the RETRAN computer code; however, this analysis was performed assuming that the PORV opened instead of the PSV. The staff stated that assuming the opening of the PORV is acceptable, because the PSV is differently qualified, and reseats mechanically. An additional independent fault would be required to cause the safety valve to fail to close. The analysis indicated that the pressurizer would fill within about 240 seconds. The licensee stated that there are multiple alarms to indicate the opening of a PORV. The licensee stated that a prompt operator action is required to close the PORV and, if the PORV does not close, the operator is to close the block valve. Because the necessary actions are prompt and simple, the NRC staff agreed that there is sufficient time to secure the inadvertently open PORV without filling the pressurizer.

On September 24, 2012, the NRC granted an EPU for St. Lucie Unit 2 that increased the authorized thermal power level about 12% to 3020 MWt. Regarding an IOECCS event, the high pressure SI pumps are incapable during power operations of delivering flow to the RCS because the pumps' shut-off head is less than the normal RCS operating pressure of 2250 psia. Therefore, the inadvertent operation of the ECCS at power event is not a credible event and was not analyzed by the licensee for the proposed EPU. The staff found that the licensee's position for not analyzing the IOECCS event to be acceptable. Regarding a CVCS malfunction, this event increases RCS inventory as an AOO that is evaluated for the effects of adding water inventory to the RCS. The NRC staff reviewed the licensee's analyses of the CVCS malfunction event and concluded that the licensee's analyses adequately accounted for operation of the plant at the proposed power level and were performed using acceptable analytical models. The staff determined that the licensee's analysis demonstrated that the pressurizer did not become water solid, assuring no water was discharged through the PSVs. Regarding an IOPORV event, the staff stated that when viewed from the mass addition perspective, this event can be evaluated in two phases: (1) an inadvertent opening of a pressurizer relief valve, followed by (2) an inadvertent ECCS actuation. In the first phase, the staff stated that this event could be mitigated by closing the open pressurizer relief valve or its block valve. If the PORV or its block valve was not closed, the staff stated that the IOPORV event would enter the second phase with actuation of the ECCS. Based on its review, the staff determined that the pressurizer overfill analysis, available alarming system, and procedures in combination with simulator exercise result had provided reasonable assurance that the pressurizer would not be expected to fill to a water solid condition that could prevent the PORV or PSV from closing after they were open, and thus, supported that the event would not generate a more serious plant conditions, meeting the third AOO acceptance criterion. The staff stated that it reviewed the licensee's analyses of the inadvertent opening of a pressurizer pressure relief valve event, and concluded that the licensee's analyses adequately accounted for operation of the plant at the proposed power level and were performed using acceptable analytical models. The staff concluded that the licensee demonstrated that the all AOO acceptance criteria are satisfactorily met.

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

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

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

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

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

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

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

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

From:	Correia, Richard
Sent:	Thursday, August 11, 2016 5:54 PM
To:	Holahan, Gary; West, Steven; Scarbrough, Thomas; Spencer, Michael; Clark, Theresa
Cc:	Weber, Michael; Hackett, Edwin; Thaggard, Mark; Coyne, Kevin
Subject:	Memorandum From: V. McCree to G. Holahan re: Charter for Backfit Appeal Review Panel Associated with Byron and Braidwood Compliance with 10 CFR 50.34(B), GDC 15, GDC 21, GDC 29, and the Licensing Basis
Attachments:	Assessment of Byron Backfit Core Damage Frequency Aug 11 2016 (final draft).docx
Importance:	High

Gary et al.,

The attached risk assessment report addresses the Byron/Braidwood backit issue. The conclusion is that the maximum benefit from a potential backfit remedy would provide a very small reduction in risk (i.e., less than 1E-06/year). It should be noted that the analysis contained in the report was narrowly focused on the backfit question under review by the Appeal Review Board and is intended to provide additional context and insights to the Board. As such, other applications of this information may not be appropriate unless this limitation is recognized.

Please let me know if you need any additional information or if you would like a briefing.

Regards

Rich

Richard P. Correia, P.E.
Director, Division of Risk Analysis
Office of Nuclear Regulatory Research
U.S. NRC

An Assessment of Core Damage Frequency

For Byron/Braidwood Nuclear Power Plants

Supporting Backfit Appeal Review Panel

ML16214A199

Prepared by RES/DRA

Selim Sancaktar, Don Helton, James Chang, John Lane

Robert Buell (INL), John Schroeder (INL)

August 11, 2016

Contents

Acknowledgement

It is imperative to mention the substantial contribution by Kevin Coyne to this project, technical, inquisitive and otherwise.

Executive Summary

By memo dated June 22, 2016, the Executive Director for Operations (EDO) established a Charter for a Backfit Appeal Review Panel to assess several issues associated with a backfit determination for the Byron and Braidwood sites (ML16173A311). The NRC staff had previously determined that the Byron and Braidwood sites were not in compliance with 10 CFR 50.34(b) because certain condition II events (faults of moderate frequency) could result in water relief through unqualified pressurizer safety relief valves (SRVs) that may allow the event to lead to a more serious condition. The EDO's Backfit Appeal Review Panel charter specifically requested information on the contribution to overall plant risk of the current configuration at Braidwood and Byron. This document provides a risk analysis intended to support resolution of this charter item. The analysis contained in this report was narrowly focused on the backfit question under review by the Appeal Review Board and is intended to provide additional context and insights to the Board. As such, other applications of this information may not be appropriate unless this limitation is recognized.

This report provides a calculation of plant core damage frequency (CDF) for the representative unit (Byron Unit 1) for plant configurations where the pressurizer pilot operated relief valve (PORV) block valves may be in different open or closed position during power operation, to provide additional information to decision makers. Additionally, the potential risk benefit of meeting regulatory requirements is estimated to provide additional information to the Backfit Appeal Review Panel.

The Byron Unit 1 SPAR model is used to analyze the plant CDF. It is deemed that the risk analysis results (plant CDF for internal events at power) for Byron Unit 1 reasonably represent those of Byron Unit 2, Braidwood Unit 1 and Braidwood Unit 2. The Byron and Braidwood plants are similar and, in fact, share the same UFSAR. The known minor differences between the units are not considered to significantly impact the results for this analysis.

An enhanced SPAR model has been used for this analysis to make plant CDF estimates for internal events at power. The analysis is supplemented by thermal hydraulic estimates (for time windows) and specific HRA analysis for key operator actions, both performed specifically for this report.

Two cases as defined below are studied and their CDs are calculated:

- CASE-A - plant as operated, assuming realistic values for human actions.

- CASE-B - plant with a "perfect backfit" that will always prevent pressurizer overfill and a subsequent challenge to the SRVs. This is modeled by assuming operator actions to unblock a blocked PORV and terminate safety injection are always successful.

The base CDF (Case-A) is 1.4E-05/year. The CDF difference between Cases A and B (CDF_A - CDF_B) is calculated to be 1.5E-07/year. This is a measure of maximum benefit that may be attained with a "perfect" backfit that avoids the issue (e.g. consequential small LOCA due to SRV failure after pressurizer overfill). This value is below the "very small" CDF delta risk threshold of 1E-06/year.[1]

It is recommended that this delta CDF be considered as the "best estimate" benefit from a "perfect" backfit that would reduce the pressurizer overfill and stuck open SRV concern. It should be also noted 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. The actual risk benefit would be lower.

An additional insight gained from this analysis is the plant risk impact of the closure of both pressurizer PORV block valves during operation. The plant CDF increases by approximately 40% when the PORV configuration goes from both valves unblocked to both PORV valves blocked. In addition, when no PORV relief path is available (e.g., both PORV valves blocked and operator actions to unblock them fail) during an inadvertent safety injection event, the pressurizer would become water solid and the SRVs challenged after approximately 20 minutes. However, when a PORV relief path is available (and the PORV is assumed capable of passing water), pressurizer pressure can be maintained below the SRV lift setpoint with full safety injection flow. Although it is assumed that the PORV would eventually fail due to repetitive cycling (either in the fully open or closed position), this provides more time for operator action to terminate pressurizer overfill and avoid a safety relief challenge. Therefore, the availability of a PORV relief path provides a longer time window for operator action to terminate injection flow and therefore reduces the human error probability of operator actions to terminate injection flow.

[1] Regulatory Guide 1.174, An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis, Revision 2 - May 2011.

An Assessment of CDF for Byron/Braidwood NPPs

1. Introduction and Objective

By memo dated June 22, 2016, the Executive Director for Operations (EDO) established a Charter for a Backfit Appeal Review Panel to assess several issues associated with a backfit determination for the Byron and Braidwood sites (ML16173A311). The NRC staff had previously determined that the Byron and Braidwood sites were not in compliance with 10 CFR 50.34(b) because certain condition II events (faults of moderate frequency) could result in water relief through unqualified pressurizer safety relief valves (SRVs) that may allow the event to lead to a more serious condition. The EDO's Backfit Appeal Review Panel charter specifically requested information on the contribution to overall plant risk of the current configuration at Braidwood and Byron. This document provides a risk analysis intended to support resolution of this charter item.

The following paragraph taken from the NRC's reaffirmation of the backfit decision (ML16095A204) summarizes the original NRC concern and assessment of the non-compliance:

> "..The analyses in the Braidwood and Byron Updated Final Safety Analysis Report (UFSAR) Sections 15.5.1, 15.5.2, and 15.6.1 are required to show that Condition II events will not cause a more serious event. This is not the case and thus, the UFSAR does not demonstrate compliance with GDCs 15, 21, and 29 and the plant-specific design basis with respect to progression of Condition II events. The UFSAR analyses of reactor coolant system mass addition (Condition II) events predict water relief through pressurizer relief valves that are not water qualified, which could result in a relief valve sticking open and causing a small break loss of coolant accident (Condition III event). Thus, Braidwood and Byron are not in compliance with 10 CFR 50.34(b). The NRC erred in approving a sequence of events that allowed the inadvertent operation of the emergency core cooling system, chemical and volume control system malfunction, and inadvertent opening of a pressurizer safety or relief valve analyses in the 2001 and 2004 Safety Evaluations (ADAMS Accession Nos. ML011420274 and ML042250516, respectively) to credit water relief through pressurizer safety valves (PSVs) that were not water qualified. The NRC has consistently applied the prohibition of progression of Condition II events, and the 2001 and 2004 approvals occurred because the NRC staff understood the PSVs to be qualified for water relief when, in fact, they were not. The licensee must take action to resolve the non-compliance, and there are a number of regulatory options available"[2]

This report provides a calculation of plant core damage frequency (CDF) for the representative unit (Byron Unit 1) for plant configurations where the pressurizer pilot operated relief valve (PORV) block valves may be in different open or closed position during power operation, to

---

[2] Byron-Braidwood UFSAR Section 15 defines inadvertent opening of a pressurizer safety or relief valve as a Condition II event. Since PRA models do not examine initiating events and consequential events in terms of "Condition-N" events, this report deliberately minimizes the use of "Condition-N' nomenclature in its plant risk modeling and assessment. However, Appendix 2 provides a description of the "Condition-N" events for Byron and Braidwood.

provide additional information to decision makers. Additionally, the potential risk benefit of meeting regulatory requirements is estimated to provide additional information to the Backfit Appeal Review Panel. The analysis contained in this report was narrowly focused on the backfit question under review by the Appeal Review Board and is intended to provide additional context and insights to the Board. As such, other applications of this information may not be appropriate unless this limitation is recognized.

Section 2 discusses the methods and risk model used to perform the analyses. Section 3 applies the method to base plant configurations and other conditions of interest. Section 4 discusses results and insights. Supporting technical discussions and calculations made for this effort are summarized in the Appendices.

2. Modeling Information and Assumptions

2.1 Method and Model

The NRC's stated assessment is that Byron/Braidwood UFSAR analyses of certain faults of moderate frequency (Condition II) include cases where water relief through unqualified pressurizer safety relief valves (SRVs, a.k.a., PSVs) could result in a relief valve failing open and causing a more serious small break loss of coolant accident (SLOCA). The specific initiating events of interest include the following:

- Inadvertent operation of the emergency core cooling system during power operation (UFSAR Subsection 15.5.1),
- Chemical and volume control system malfunction that increases reactor coolant inventory (UFSAR Subsection 15.5.2),
- Inadvertent opening of a pressurizer safety or relief valve (UFSAR Section 15.6.1).

However, if either of the pressurizer PORV flow paths are available (i.e., both the PORV and its block valve are capable of opening), then the SRV challenge could be prevented if safety injection is terminated in a timely manner. The PORVs and their associated flow paths can provide an acceptable relief path for a Condition II event[3] until the Pressurizer Relief Tank (PRT) is filled[4]. The PORVs can be opened either by a proceduralized operator action, or, if the PORVS are aligned for automatic operation, by pressurizer pressure reaching PORV setpoint(s), which are lower than those of SRVs. During operation, PORV block valves can be closed to support routine testing and maintenance or to isolate an inoperable PORV. Therefore, this analysis assesses the impact on a unit's CDF for internal events due to a unit operating with different PORV block valve configurations (e.g. closed or open during power operation). If at least one block valve is open

[3] The relief capacity of a single PORV at full RCS pressure was estimated to be approximately 40% greater than the injection rate from two high pressure emergency core cooling pumps. If it is assumed that the SRV lifting setpoint is unaffected by the presence of water due to pressurizer overfill and decay heat removal capability is maintained, the flow through a single PORV is sufficient to prevent an SRV challenge.

[4] Once the PRT rupture disk fails, the event becomes more serious.

and no additional component or human failures are postulated (including timely operator action to address rising pressurizer level), then pressurizer overfill (that results in passing water through SRVs) would not be expected during an inadvertent safety injection event. For some other scenarios (where the RCS makeup rate is lower), operation of the PORVs may be sufficient to prevent a pressure challenge to the SRVs. However, dependence on the continued successful operation of PORV relief valve(s) during a prolonged overfill event may not be reliable since repeated cycling (opening and closing) of the PORVs may lead to their failure.

The Byron Unit 1 SPAR model is used to analyze the plant CDF. It is deemed that the risk analysis results (plant CDF for internal events at power) for Byron Unit 1 reasonably represent those of Byron Unit 2, Braidwood Unit 1 and Braidwood Unit 2. The Byron and Braidwood plants are similar and, in fact, share the same UFSAR. The known minor differences between the units[5] are not considered to significantly impact the results for this analysis.

Since spurious Safety Injection (SI) was previously subsumed within the General Transients initiating event in the SPAR model, proceduralized operator actions are introduced into the SPAR model for opening of closed block valves, and termination of SI (prior to pressurizer overfill). The human error probabilities of such operator actions are estimated based on contemporary HRA methods (see Appendix 2).

Three basic plant operation cases based on the position of the two PORV block valves during power operation are defined:

Case-1	Both block valves A and B are open. This is the most favorable case from a CDF point of view.
Case-2	One block valve is closed, the other is open.
Case-3	Both block valves are closed. This is the least favorable case from a CDF point of view.

There is an implied assumption here that the PORV is functional (i.e., the PORV block valve is closed for reasons other than a failed[6] PORV).

---

[5] For instance, Byron and Braidwood Units 1 use BWI steam generators, while Unit 2 at both sites use Westinghouse D5 steam generators; thus there are differing primary-side and secondary-side operational and off-normal set-points between Unit 1 and Unit 2 at each site. The sites also obviously have different site characterizations.

[6] [6] In this context, a failed PORV is considered to be a PORV that is not capable of performing its pressure relief function when unblocked. For this analysis, when the block valve is closed, it is considered to be for conditions when the PORV is still available to perform a pressure relief function (e.g., the block valve is closed to mitigate minor PORV seat leakage).

2.2 Success Criteria

A summary of key success criteria used for pressurizer overfill potential and termination of SI injection in applicable event trees is given below.

Success Criteria -1	If at least 1 pressurizer PORV path is open (both PORV and its block valve), SRVs will not pass water upon spurious actuation of SI (both SI pumps + both charging pumps). Note that the PORV is assumed to eventually fail after repeated open/shut cycles and/or the pressurizer relief tank will overfill requiring the termination of injection (see also success criteria 3)
Success Criteria -2	If operators terminate SI before the calculated time windows for spurious safety injection, steam line break, or SGTR, an RCS LOCA due to a failed SRVs (due to passing water) will not occur. As discussed in Appendices 1 and 2, the time window for operator action under these conditions is assumed to be 40 minutes.
Success Criteria-3	In spurious SI and SLB events, SI must eventually be stopped (or throttled) in a timely manner to prevent pressurizer overfill, rupture disc failure for the pressurizer relief tank, or PORV failure even if a pressurizer PORV path is initially open.[7]

SLOCA definition in SPAR Model	between 3/8 in. and 2 in	Exceeds normal charging flow. Normal charging cannot maintain pressurizer level.
# of charging and SI pumps actuated upon ESF signal	2 charging and 2 SI pumps	Total flow from these pumps will determine the time to PORV/SRV challenge, etc.

2.3 Pressurizer Overfill and SI Termination

In spurious SI and steamline break events, safety injection must be stopped in accordance with procedures in a timely manner to prevent pressurizer overfill, which is postulated to cause SRV failure and a small LOCA. Moreover, in some CVCS malfunction events where the pressurizer level is increasing, the operators must stop or control charging flow and letdown to terminate the event to prevent pressurizer overfill which will cause SRV LOCA.

These three situations are discussed below as a part of assessment of risk for an initiating event leading to a consequential SLOCA. The events themselves are described in greater detail in Appendix 4.

---

[7] The reason that long-term reliance on PORVs (as opposed to terminating SI) is unacceptable is because repeated cycling of the PORVs increases the probability that they will fail. If it is assumed that the repeated PORV cycling would ultimately lead to them failing closed and no other leakage is present, the CCPs are capable of lifting the SRVs at Byron/Braidwood. If instead the PORVs fail open after repeated cycling, a SLOCA is initiated. Thus, this success criteria is used in not accepting long-term reliance on the PORVs (and thus requiring that SI be terminated in some reasonable timeframe).

2.3.1 Consequential LOCAs and Spurious SI Event

In SPAR models, like in other PRA models, an initiating event may develop into another more consequential event due to additional component failures or failure of operator actions. In such cases, the affected accident sequences in the event tree are transferred to the more consequential event tree. In simpler cases, such sequences are resolved within the event tree. Examples of such transfers are consequential LOCAs (PORV/SRV LOCAS, reactor coolant pump (RCP) seal LOCAs), consequential steam generator tube ruptured (SGTRs), consequential LOOP, and anticipated transients without SCRAM (ATWS).

Development of relatively benign and common initiating events into more serious (consequential) events is of interest and is further discussed here. The first candidate condition II initiating event in which a consequential PORV/SRV LOCA may develop due to additional failures or plant condition (e.g. initially-closed PORV block valves) is the spurious SI actuation event (IE-ISINJ). This event tree is further discussed in Appendix 4.

2.3.2 Steamline Break Events

In a SLB event, an SI signal is produced due to shrinkage of RCS inventory, thus lowering of the pressurizer level. However, there is no actual RCS inventory loss due to this initiating event. An SI signal will occur based on low steamline pressure, the pressurizer level will rise and, unless SI is stopped or controlled, the pressurizer will overfill. The, SRVs will then pass water and a SLOCA will be postulated, as in the ISINJ event.

SI termination in SLB event trees is routinely modeled. If it fails the accident sequence is transferred to SLOCA. For completeness of CDF assessment, two SLB ETs are added to the SPAR model. Thus, contribution of potential consequential LOCAs from SLB events inside and outside of containment are included in the CDF assessment.

2.3.3 CVCS Malfunctions

The operators respond to CVCS malfunctions that could increase the water level in the pressurizer by using Abnormal Operating Procedures (AOPs). If the event is not terminated and the pressurizer level continues to increase, a reactor trip is generated. This event is subsumed in the Transients Initiating event category and is not explicitly modeled in PRAs.

The frequency of "*CVCS malfunction AND failure to terminate it before reactor trip'* is not readily available. In addition, as discussed in Appendix A, there is expected to be at least 50 minutes available before pressurizer overfill during a CVCS malfunction event. Therefore, in the judgment of the authors, the contribution of this event to plant risk is small and would not affect the insights; thus it is not further pursued in this study.

2.3.4 Other SRV Failures

Random failures of SRVs to open prematurely (spurious opening), and fail to close when needed (fail to reclose), are not further studied in this report since they are deemed to be lesser contributors to plant risk, compared to the failure modes included in this study (e.g. failure due to passing of water after pressurizer overfill). The base SPAR model has SRVs fail to open and also fail to reclose modes included in response to PORV paths failing. The success criterion used is that if SRVs open and reclose, then consequential SLOCA is avoided. A sensitivity analysis (setting SRV failure to reclose basic event to failure) using this model indicated that the delta CDF was insignificant (e.g. third significant figure change in CDF). Thus, this was not further pursued.

It is believed that further pursuing these types of additional failure modes would not provide new insights, and would also further complicate the Boolean logic of the model and would require additional data research, which is beyond the scope of this study.[8]

[8] The base model was run with the change set that sets the 3 "SRV Fail to Reclose" basic events set to TRUE; the CDF change was in the third significant figure. Spurious opening of SRVs are not included in the existing SLOCA frequency. Although there is insufficient data available in the Operating Experience database to calculate an initiating event frequency for spurious SRV opening with high confidence, based on small number of operating events, the risk contribution from a spurious SRV opening is considered to be negligible. SRV leakage due to pressurizer overfill with conservative success criteria already provides a non-insignificant delta CDF and is expected to bound the small risk contribution from spurious openings. See also Appendix 4, Section D.3 for "Notes on SRV Failures" For additional details.

3. CDF of Plant Configurations

The revised SPAR model uses information from the last benchmarking activity of the Byron/Braidwood SPAR models (i.e., comparison with the utility PRA models) to arrive at a probability of one PORV block valve being closed during power operation of 0.0633[9]. In addition, we will postulate that the plant would be in a condition with both PORV block valves closed 1% of the time (assuming random occurrence of this with the above values would give a probability of 0.004). Thus we will use the following plant conditions to estimate the CDF:

87% of the time, the plant will be operating with both block valves open;
12% of the time, the plant will be operating with one block valve closed, the other open;
1% of the time, the plant will be operating with both block valves closed.

This information was used in the Byron SPAR Model to quantify the plant CDF for internal events, and identify failure combinations (cutsets) containing random failures, common cause failures, human errors, test and maintenance unavailabilites, etc. that lead to core damage.

Table 3.1.1-1 summarizes the CDF of the basic plant configurations (cases) for the base case (current Byron/Braidwood configuration):

Table 3.1.1-1 Plant CDF (internal events) for Basic Plant Configurations (Cases)		
Case	**Description**	**CDF**
Case-0	Both block valves are open	1.36E-05
Case-1	One block valve is open, the other is closed	1.78E-05
Case-2	Both block valves are closed	1.89E-05

The block valve configuration percentages presented in Section 2.3 are used along with these CDF results to arrive at the base plant CDF (for internal events):

CDF = 0.87 * 1.36E-05 + 0.12 * 1.78E-05 + 0.01 * 1.89E-05
CDF = 1.41E-05 per year of power operation (8760 hours).

Table 3.1.1-2 provides the CDF results by initiating event categories.

[9] The base SPAR model uses a similar value of 0.05.

Table 3.1.1-2 Plant CDF by Initiating Events

			Block Valve Configurations (see Table 3.1.1-1)			
	IE Name	**Description**	**IE Frequency**	**Case-0 CDF**	**Case-1 CDF**	**Case-2 CDF**
1	IE-DLOESW	LOSS OF ESW AT BOTH UNITS (1 AND 2)	2.19E-04	4.14E-06	4.14E-06	4.76E-06
2	IE-ISINJ	INADVERTENT SAFETY INJECTION	1.47E-02	1.27E-07	2.01E-07	2.58E-07
3	IE-ISL-HPI	ISLOCA IE 2-CKV HPI INTERFACE	2.18E-06	3.05E-09	3.05E-09	3.05E-09
4	IE-ISL-LPI	ISLOCA IE 2-CKV LPI INTERFACE	2.18E-06	3.04E-08	3.04E-08	3.04E-08
5	IE-ISL-RHR	RHR PIPE RUPTURES	3.76E-06	5.09E-07	5.09E-07	5.09E-07
6	IE-LDCA	LOSS OF DC BUS 111	3.69E-04	1.39E-07	1.41E-07	1.50E-07
7	IE-LDCB	LOSS OF DC BUS 112	3.69E-04	6.31E-09	8.28E-09	8.85E-09
8	IE-LLOCA	LARGE LOCA	2.50E-06	5.09E-08	5.09E-08	5.09E-08
9	IE-LOAC141	LOSS OF AC BUS 141	3.34E-03	1.61E-06	1.61E-06	1.69E-06
10	IE-LOAC142	LOSS OF AC BUS 142	3.34E-03	7.09E-07	7.10E-07	7.15E-07
11	IE-LOCCW	LOSS OF COMPONENT COOLING WATER	2.46E-04	2.65E-07	2.67E-07	2.93E-07
12	IE-LOCHS	LOSS OF CONDENSER HEAT SINK	5.86E-02	9.70E-07	1.26E-06	1.33E-06
13	IE-LOESW	LOSS OF ESSENTIAL SERVICE WATER	2.46E-04	1.28E-07	1.28E-07	1.36E-07
14	IE-LOMFW	LOSS OF MAIN FEED WATER	6.89E-02	1.13E-06	1.48E-06	1.56E-06
15	IE-LONSW	LOSS OF NON-ESSENTIAL SERVICE WATER	2.46E-04	3.96E-09	5.20E-09	5.46E-09
16	IE-LOOPGR	GRID RELATED LOOP	1.22E-02	3.58E-07	3.73E-07	4.04E-07
17	IE-LOOPPC	PLANT CENTERED LOOP	1.93E-03	1.03E-08	1.26E-08	1.45E-08
18	IE-LOOPSC	SWITCHYARD RELATED LOOP	1.04E-02	1.22E-07	1.34E-07	1.50E-07
19	IE-LOOPWR	WEATHER RELATED LOOP	3.91E-03	1.85E-07	1.90E-07	2.02E-07
20	IE-MLOCA	MEDIUM LOCA	1.50E-04	6.63E-07	6.63E-07	6.63E-07
21	IE-SGTR	SG TUBE RUPTURE	2.07E-03	9.34E-08	9.34E-08	9.36E-08
22	IE-SLBIC	STEAM LINE BREAK INSIDE CONTAINMENT	3.12E-04	6.31E-09	7.88E-09	1.11E-08
23	IE-SLBOC	STEAM LINE BREAK OUTSIDE CONTAINMENT	6.55E-03	1.24E-07	1.57E-07	2.27E-07
24	IE-SLOCA	SMALL LOCA	3.67E-04	1.33E-06	1.33E-06	1.33E-06
25	IE-TRANS	TRANSIENT	6.90E-01	7.62E-07	4.21E-06	4.24E-06
27	IE-XLOCA	REACTOR PRESSURE VESSEL RUPTURE	1.00E-07	1.00E-07	1.00E-07	1.00E-07
		Total =	**8.78E-01**	1.36E-05	1.78E-05	1.89E-05

4. Results and Insights

An enhanced SPAR model has been used for this analysis to obtain plant CDF estimates for internal events at power. The analysis is supplemented by thermal hydraulic estimates (for time windows) as given in Appendix 1, and specific HRA analysis for key operator actions, as given in Appendix 2. In this section, three scenarios labeled as Case-A, Case-B and Case-C are analyzed:

- CASE-A plant as operated, assuming realistic values for human actions.
- CASE-B plant with a "perfect backfit" that will always prevent pressurizer overfill and a subsequent challenge to the SRVs. This is modeled by assuming operator actions to unblock a blocked PORV and terminate safety injection are always successful.

The human error probability (HEP) values of the relevant operator actions for the cases studied are shown in Table 4-1. The plant CDF for all internal events for each case is calculated. The CDF results are summarized in Table 4-2.

It should be noted that, although the issue at hand focuses on spurious SI events and CVCS malfunctions, "termination of SI injection" appears routinely in SLB and SGTR events; these events are included in the PRA model used for this analysis.

Plant as Operated (CASE-A)

In Section 3, the plant operating condition mix of

87% of the time, the plant will be operating with both block valves open;
12% of the time, the plant will be operating with one block valve closed, the other open;
1% of the time, the plant will be operating with both block valves closed;

is used to estimate the average internal events CDF as 1.41E-05 for a year of power operation (8760 hours). The results are provided in Table 4-2.

Plant with Perfect Overfill Protection (CASE-B)

This case is simulated by setting the operator action failure probability of "Terminate SI Injection" to zero. The results are presented in Table 4-2.

The average CDF difference between the two cases is the maximum expected CDF reduction if a "perfect" backfit is implemented. Using the assumptions stated in Sections 2 and 3, the difference in CDF is 1.5E-07/year. It should be also noted that any backfit remedy is not expected to be perfectly effective. Therefore, this delta represents the maximum possible benefit from any backfit plant change. Actual risk benefit would be lower.

An additional insight gained from this analysis is the plant risk impact of the closure of both pressurizer PORV block valves during operation. As shown in Table 4-2, the plant CDF increases by approximately 40% when the PORV configuration goes from both valves unblocked to both PORV valves blocked. In addition, when no PORV relief path is available (e.g., both PORV valves blocked and operator actions to unblock them fail) during an inadvertent safety injection event, the pressurizer would become water solid and the SRVs challenged after approximately 20 minutes. However, when a PORV relief path is available (and the PORV is assumed capable of passing water), pressurizer pressure can be maintained below the SRV lift setpoint with full safety injection flow. Although it is assumed that the PORV would eventually fail due to repetitive cycling (either in the fully open or closed position), this provides more time for operator action to terminate pressurizer overfill and avoid a safety relief challenge. Therefore, the availability of a PORV relief path provides a longer time window for operator action to terminate injection flow and therefore reduces the human error probability of operator actions to terminate injection flow.

Table 4-1 HEP Values of Relevant Operator Actions in the Cases Studied

Alias (*)	Basic Event Name	Basic Event Description	Case-A HEP	Case-B HEP
OPA-1	HPI-XHE-XM-THRTL1	OPERATOR FAILS TO CONTROL/TERMINATE SAFETY INJECTION FLOW (OPA-1 - ISINJ)	1.00E-03	0
OPA-2	HPI-XHE-XM-THRTL2	OPERATOR FAILS TO CONTROL/TERMINATE SAFETY INJECTION FLOW (OPA-2 - SLBOC)	3.73E-03	0
OPA-3	PPR-XHE-XM-UNBLOCK	OPERATOR FAILS TO OPEN PORV BLOCK VALVES PER OPA-3	2.25E-03	0
OPA-4	HPI-XHE-XM-THRTL1-DEP	OPERATOR FAILS TO CONTROL/TERMINATE SAFETY INJECTION FLOW (OPA-1 after OPA-3 fails)	5.01E-01	0
OPA-5	HPI-XHE-XM-THRTL3	OPERATOR FAILS TO CONTROL/TERMINATE SAFETY INJECTION FLOW (OPA-2 - SLBIC)	1.25E-03	0
(**)	HPI-XHE-XM-THRTL	OPERATOR FAILS TO CONTROL/TERMINATE SAFETY INJECTION FLOW (SGTR)	2.00E-03	0

(*) Aliases are introduced to avoid using long basic event names in discussions in the report sections.
(**) This HEP was already in the SPAR model; the others in the table are analyzed in Appendix 2.

Table 4-2. Change in CDF with Overfill Protection

Case	Description	Both block valves open during operation	one block valve open during operation	Both block valves closed during operation	Average (*)
		CDF0	CDF1	CDF2	CDF
A	Plant as operated	1.358E-05	1.781E-05	1.893E-05	1.41E-05
B	Perfect overfill protection	1.343E-05	1.766E-05	1.850E-05	1.40E-05
A-B	Delta CDF	1.5E-07	1.5E-07	4.3E-07	**1.5E-07**

(*) average CDF = 0.87* CDF0 + 0.12* CDF1 + 0.01* CDF2

Note: Calculations performed using a greater number of significant digits than shown in the table.

5. References

1. BYRON SPAR Model 8.27, modified for this project by addition of 4 ETs.
2. 2BEP-0, Reactor Trip or Safety Injection, Rev. 201 WOG 2
3. 1BEP-ES-1.1 SI Termination, Rev. 200, WOG 2
4. 2BEP-2 Faulted Steam Generator Isolation, Rev. 200, WOG 2
5. SPAR-H Step-by-Step Guidance, INL/EXT-10-18533, Revision 2, May 2011.

Appendix 1. Event/Procedural Narratives

A.1-1 Inadvertent Safety Injection (ISINJ)

Initial conditions and initial fault: Unit operating at 100% power when an instrumentation failure (or other spurious cause) results in an SI signal (both channels).

Subsequent plant response: The safety injection (SI) signal will cause the reactor protection system to send a reactor trip signal[10], resulting in reactor trip. Pressurizer level will shrink, and pressurizer pressure will drop, due to this trip. The SI signal will also isolate letdown, and cause the charging pumps to automatically realign from their normal charging alignment (drawing from the VCT and injecting through the normal charging line) to their ECCS injection alignment (drawing from the RWST and injecting through the SI injection lines). The turbine and main feedwater pumps will trip, and auxiliary feedwater will initiate and maintain SG levels at their post-trip setpoint[11]. Condenser steam dumps (a.k.a., turbine bypass valves) will modulate in an attempt to maintain no-load T_{ave} on the primary side. ECCS will begin injecting: both centrifugal charging pumps will actually inject, while SI pumps will be dead-headed. RHR pumps and cold-leg accumulators will also be dead-headed. The mismatch between injection and letdown will cause the pressurizer level to increase, which will compress the steam bubble in the pressurizer and cause primary-side pressure to increase. The addition of the cold RWST water will tend to decrease T_{ave}. Pressurizer sprays will actuate to counteract the pressure increase, but this does not change the mass imbalance since the water is being pulled from one of the cold legs. Barring operator action, the pressure will increase to the point that a PORV or SRV (depending on the position of the PORV block valves) will lift on steam relief. Subsequently, the pressurizer will overfill and the PORVs/SRVs will pass water.

Procedural path:

For the initial procedural narrative, it is assumed that no subsequent failures (beyond the inadvertent SI) occur. At a high-level, the operator response to this event will be to:

- Ensure that the SI was not caused by a main steam line break, primary-side pipe break, or SG tube rupture;
- Determine that ECCS is not required, and
- Secure ECCS.

More specifically, upon the SI and reactor trip, the operators will enter E-0 [1BEP-0 at Byron Unit 1[12]], "Reactor Trip or Safety Injection." Early steps in this procedure will involve commencing evaluation of Emergency Plan conditions, verifying reactor and turbine trip[13], verifying power to 4kV ESF busses, and checking SI status. At this point a 9-page balance-of-plant verification will be initiated in parallel to continuing with E-0. Subsequent E-0 steps will involve verifying ECCS

[10] Note that there are ECCS malfunctions that may not cause a reactor trip, but these are not considered further in this description. They are considered in the PRA modeling, in that RPS success is queried, and failure leads to transfer to the ATWS event tree.

[11] Failure of AFW is also not further considered in this plant response description, but is considered in the PRA modeling, in that failure of AFW results in querying of FR-H.1 actions (primary-side feed and bleed).

[12] The readily available version of this procedure was Revision 201 (Circa 2010).

[13] If reactor trip indications are not satisfied, operators are directed to FR-S.1 (Response to Nuclear Power Generation/ATWS).

pump actuation, verifying fan coolers have transferred to accident mode, verifying that containment isolation Phase A and containment ventilation isolation have occurred, verifying AFW operation, verifying ECCS pump operation, and determining if main steam isolation is required (in this case it would not be; whereas for a steam line break automatic closure would be expected). E-0 would go on to direct the operators to check the need for containment sprays (not needed in this case), verify AFW flow[14], verify ECCS valve alignment, and check pressurizer PORVs and spray valves. Step 18 of E-0 is the first place where operators are directed to close, and if necessary block, a PORV if it appears to have failed open (pressure less than 2315 psig and PORV indicates open). This is also the step when operators are directed to open a PORV block valve if both PORVs are blocked and either is available (un-failed). If a PORV is failed-open and cannot be blocked, the operators are directed to transition to E-1 [1BEP-1 at Byron Unit 1], "Loss of Reactor or Secondary Coolant." If PORVs are behaving normally, operators will proceed to checking pressurizer spray status. Next operators will verify that RCS temperature is at or trending to no-load T_{ave} (557F for Byron Unit 1), and check whether RCS conditions prompt stopping the RCPs (for this event, they won't). Next operators will confirm that SG secondary pressure boundaries are intact, that SG tubes appear intact, and that there are no indications of a primary-side LOCA. Having found no indications of these conditions, at Step 24 of E-0 operators are directed to check if ECCS flow should be reduced, based on:

- Acceptable RCS subcooling;
- Sufficient secondary-side heat removal capability;
- RCS pressure stable or rising; and
- Pressurizer level greater than 12%.

For an uncomplicated inadvertent SI event, the above conditions would prompt operators to transfer to ES-1.1 [1BEP ES-1.1 for Byron Unit 1[15]], "SI Termination." (Note that the critical safety function status trees are not invoked on this procedure path.)

Steps 1-3 of ES-1.1 direct operators to reset SI and containment isolation, and to establish instrument air to containment. Step 4 directs operators to stop all but one charging pump. Depending on the plant response to reducing charging flow, operators will either be directed to transfer to ES-1.2 [1BEP ES-1.2 at Byron Unit 1], "Post LOCA Cooldown and Depressurization" (if RCS pressure is dropping), or remain in ES-1.1 (otherwise). If remaining in ES-1.1, operators are directed (Steps 6-9) to terminate high-head SI (centrifugal charging and SI pumps) and re-establish normal charging flow (centrifugal charging pumps). Subsequent steps in ES-1.1 are focused on stopping RHR pumps, re-establishing letdown, and generally preparing the plant for entry in to 1BGP 100-5, "Plant Shutdown and Cooldown."

A graphical representation of this procedure path is provided in Figure A1-1.

Effect of general post-trip complications: Any number of complications car arise following the inadvertent SI, such as:

- Pre-existing plant conditions (e.g.., equipment out-of-service)
- Operator error
- SI/trip-induced loss-of-offsite power (a.k.a., LOCA-LOOP)

---

[14] If AFW flow cannot be established, and SG levels in all SGs have dropped below 10% narrow range level, Step 15 will direct the operators to enter FR-H.1 (Response to Loss of Secondary Heat Sink).

[15] The readily available version of this procedure was Revision 200 (Circa 2010).

- Partial SI actuation
- Failure-to-start of charging pump(s)
- Failure of fan coolers to transfer to accident mode
- Failure of containment isolation Phase A or containment ventilation isolation
- Failure of AFW
- MSIV closure or containment spray actuation (unexpected)
- PORV fails open
- Failed PORV cannot be blocked closed
- Etc.

Some of these failures would lead to a substantively similar procedural path (e.g., failure of fan coolers to transition to accident mode would prompt additional manual actions but no procedural diversion), while other failures would lead to notably different procedural paths (e.g., a relief valve being stuck-open and unisolable would lead to a LOCA-response procedural path). For a real-world example of a complicated response to an inadvertent SI, see Attachment 2 of "Millstone Power Station Unit 3 – NRC Special Inspection Report 05000423/2005012 [ML051860338].

Timing considerations:

A notional early-event plant response timeline is presented in Table A1-1. These time estimates were generated based on consideration of three sources: (i) hand calculations using readily-available Byron design/operation information, including dated information designated as 'For Training Purposes Only,' (ii) the aforementioned Millstone event timeline, and (iii) a side calculation using the Byron Unit 1 (revision 8) MELCOR model and MELCOR v2.1.7044. These estimates reflect a situation where at least 1 PORV is available (and operates), no relief valves fail open during their early operation, and both trains of ECCS actuate and inject. Based on this information, it is estimated that the operators have roughly 10 minutes to secure charging pumps in order to avoid PORV cycling, and roughly 15-20 minutes to avoid passing water through the PORVs. If the PORVs are not available, the SRVs would be expected to begin passing water at approximately 20 minutes. In the case that only one train of ECCS actuates and injects, these values would be roughly double the stated values. A range is given for the latter value because the methods for generating these timing estimates did not consider the potential for water entrainment as the relief valves cycle with a nearly full pressurizer.

The time estimated (~20 minutes) is the time the pressurizer goes solid, and represents a minimum time for the SRV passing water (neglecting entrainment). The time at which the SRV passes water is dependent on the availability and reliability of the PORVs when passing water. To the extent that the PORV(s) provide the necessary relief path, and that the PORV(s) reclose after SI is terminated, a much longer time may be available to terminate SI. To balance the desire to not rely on sustained PORV performance under such conditions, but to acknowledge that a much longer time may be available, an alternate time of 40 minutes is proposed for the purpose of assessing HEP sensitivity to the time available.

Table A1-1. Inadvertent SI Notional Plant Response with no Operator Action

Event	Time estimate (minutes)	Notes
SI actuation	0	
Reactor trip	0	Reactor trips on SI signal
Pressurizer high-level alarm	10	MCR annunciator
Pressurizer PORV begins cycling with steam relief	10	
High pressurizer level reactor trip setpoint reached	15	Reactor is already tripped
Pressurizer level at 100% indicated level	18	
Pressurizer becomes water solid	19	If available, the PORVs are assumed to be capable of passing water and avoiding a challenge to the SRVs. If no PORV relief path is available, the SRVs would be expected to begin passing water at this time.
PORVs are assumed to fail after repeated open/close cycles	40	Even if available, repeated open/close cycles of the PORVs would be expected to eventually cause the PORV to fail either open or closed. This establishes the time window for operator action to terminate safety injection flow.

Inadvertent SI occurs

Emergency Response Guidelines

BEP-0: Reactor Trip or Safety Injection

Steps 1 – 4: commencing evaluation of Emergency Plan conditions, verifying reactor and turbine trip, verifying power to 4kV ESF busses, and checking SI status

Steps 5: 9-page balance-of-plant verification will be initiated in parallel to continuing with E-0

Steps 6 - 13: verifying ECCS pump actuation, verifying fan coolers have transferred to accident mode, verifying that the containment isolation Phase A and containment ventilation isolation, verifying AFW operation, verifying ECCS pump operation, and determining if main steam isolation is required

Steps 14-17: check the need for containment sprays, verify AFW flow, and verify ECCS valve alignment

Step 18: check pressurizer PORVs and spray valves

Steps 19-20: verify that RCS temperature is at or trending to no-load T_{ave} and check whether RCS conditions prompt stopping the RCPs

Steps 21-23: confirm that SG secondary pressure boundaries are intact, that SG tubes appear intact, and that there are no indications of a primary-side LOCA

Step 24: check if ECCS flow should be reduced

[redacted]

Note: CSFSTs are not invoked on this procedure path prior to securing ECCS. Even if they were, when PRZ level >92%, status trees will invoke FR-I.1, but the operators will be immediately kicked out because ECCS is still operating

If PORV is failed in the open position, and the block valve can not be closed, transition to BEP-1, "Loss of Reactor or Secondary Coolant."

If RCS pressure is dropping, transition to BEP ES-1.2, "Post LOCA Cooldown and Depressurization."

Figure A1-1. Notional Procedural Path for Inadvertent SI

A.1-2 Main Steam Line Break

Initial conditions and initial fault: Unit operating at 100% power when a break on the secondary-side (e.g., spurious lift of a SG relief valve, degradation-induced pipe rupture) occurs.

Subsequent plant response: The initial plant response will be fundamentally tied to the size of the secondary-side break. Here it is assumed that the break is relatively large, leading to fairly rapid depressurization of the secondary-side. Such a condition would lead to a relatively rapid:

- SI signal (steamline pressure < 614 psig),
- Reactor trip (overpower ΔT or SI-initiated), and
- MSIV closure (steamline pressure < 614 psig or steamline pressure change rate of > 165 psig over a time interval of 50 seconds).

For simplicity, it is assumed here that the above responses occur within the first minute (a reasonable assumption for large breaks). With an early reactor trip, an early SI actuation, and the faulted SG isolated, the subsequent plant response is functionally similar to the inadvertent SI previously discussed[16].

Procedural path:

Again, the presumption here will be that the steam leak is large enough to result in relatively quick secondary-side depressurization, such that reactor trip and SI occur prior to operator actions prompted by the steam leak. In such a condition, operators will enter E-0 upon reactor trip or SI. The initial pass through E-0 will be very similar to what was previously characterized for an inadvertent SI event, with notable deviations being:

- E-0 Step 13 will direct the operators to ensure MSIVs (and MSIV bypass valves) have closed, and to close them if they are not already closed (and conditions warrant);
- E-0 Step 14 will direct operators to stop all RCPs if containment sprays have actuated on high containment pressure (20 psig)[17].
- E-0 Step 21 will direct the operators to E-2, "Faulted Steam Generator Isolation," if any SG pressure is dropping in an uncontrolled manner, or any SG is completely depressurized.

As such, for a steam line break outside of the MSIV (of the size presumed here), the inadvertent SI situation previously described remains applicable, from an HRA context and sequence timing perspective (and the previously provided timings and context are assumed to apply).

However, for a steam line break inside of the MSIV, a different procedural path applies, since the operators will be directed to E-2 prior to reaching the step that would otherwise direct them to ES-1.1. E-2 [1BEP-2 at Byron Unit 1[18]], Steps 1-2 re-verify MSIV closure and a faulted SG. Step 3 has the operators identify the faulted SG. Step 4 has the operators attempt to isolate the faulted SG by closing auxiliary feedwater isolation valves, confirming main feedwater to the faulted SG

---

[16] This assertion is stated from the perspective of PRA/HRA modeling, not safety analysis. It is recognized that from a safety analysis perspective, the limiting core response (e.g., DNBR) is different for these events.

[17] With the RCPs tripped, Step 19 will direct the operators to check RCS temperatures using cold leg thermocouples rather than T_{ave}.

[18] The readily available version of this procedure was Revision 200 (Circa 2010).

is isolated, verifying that the SG PORV is closed (and blocking it if it is not), and verifying SG blowdown and blowdown sample isolation valves are closed. Steps 5-6 direct operators to monitor auxiliary feedwater pump suction pressure and check secondary radiation. Step 7 of E-2 mirrors Step 24 of E-0, in checking if conditions warrant reduction of ECCS flow. If they do, then operators are directed to transition to ES-1.1, and the procedural path then resumes that previously discussed for the inadvertent SI situation. Put more simply, a steamline break upstream of the MSIV will mirror the procedural path for a steamline break downstream of the MSIV or an inadvertent SI, except that an additional procedure and 6 extra steps apply.

Effect of general post-trip complications: The situation here is similar to that previously described for an inadvertent SI.

Timing considerations: Per the proceeding discussion, the sequence timing in Table A1-1 is also applied here.

A.1-3 Chemical and Volume Control System (CVCS) Malfunction

Initial conditions and initial fault: Unit operating at 100% power; a myriad of malfunctions can occur that will result in a myriad of different plant responses – two specific malfunctions are considered here:

1. A malfunction in CVCS or operator error causes isolation of normal letdown;
2. The controlling pressurizer level transmitter (LT) fails low.

Subsequent plant response:

Letdown isolation: In this case, a mis-match between charging and letdown will cause pressurizer level to gradually rise. If in auto, the pressurizer level controller will attempt to address the mis-match by reducing charging flow. If this does not happen, a pressurizer high-level alarm will occur at 70%. If no action is taken, a reactor trip will occur when pressurizer level reaches 92%[19]. Upon reactor trip, pressurizer volume will shrink (level will drop). If the mis-match persists and no action is taken, the pressurizer will eventually over-fill (but no SI signal would be generated unless a relief valve fails open).

Controlling pressure LT fails low: In response to this failure, charging will maximize while letdown will isolate. No immediate reactor trip or SI signal is generated. The preceding description applies, on a somewhat faster timescale.

Procedural path:

The plant-specific annunciate response procedures are not available, but would likely direct the operators (upon the high pressurizer level alarm) to investigate and address the cause of the rising pressurizer level. The plant-specific abnormal operating procedures (AOPs), as of the 2010 versions readily available, do not appear to directly address this situation:

- 1BOA ESP-2, Rev 000, RE-ESTABLISHING CV LETDOWN UNIT 1
- 1BOA PRI-1, Rev 105, EXCESSIVE PRIMARY PLANT LEAKAGE UNIT 1
- 1BOA PRI-2, Rev 104, EMERGENCY BORATION UNIT 1
- 1BOA PRI-4, Rev 102, ABNORMAL PRIMARY CHEMISTRY UNIT 1
- 1BOA PRI-12, Rev 102, UNCONTROLLED DILUTION UNIT 1
- 1BOA RCP-1, Rev 102, REACTOR COOLANT PUMP SEAL FAILURE UNIT-1
- 1BOA RCP-2, Rev 103, LOSS OF SEAL COOLING UNIT 1

Once a reactor trip occurs, then the operators will enter E-0. At Step 4, operators will be directed to ES-0.1 [1BEP ES-0.1 at Byron Unit 1[20]], "Reactor Trip Response." Step 3 of this procedure will direct the operators to check pressurizer level, which (assuming the condition has gone undetected until this point) is the next opportunity for the operators to recognize the high pressurizer level and/or the failed level transmitter. This step also directs the operators to verify that charging and letdown are in service, restore charging and letdown, and verify that pressurizer level is trending toward program level.

---

[19] This assumes that pressurizer sprays adequately control pressure as the steam bubble is squeezed; otherwise a reactor trip on high pressurizer pressure may also be relevant.

[20] The readily available version of this procedure was Revision 200 (Circa 2010).

Effect of general post-trip complications: Examples of complicating factors include:

- Pre-existing plant conditions (e.g.., equipment out-of-service)
- Operator error
- PORV/SRV fails open
- Failed PORV cannot be blocked closed

Timing considerations: Timing estimates are provided in Table A1-2 based on simple mass balance calculations. The maximum charging flow for Byron/Braidwood is not known to the authors. It is known that the high charging flow alarm setpoint is 150 gpm, and that the main control board flow readout's highest indication is equivalent to this value. Thus, for the sake of this estimate, it will be assumed that the maximum flow is somewhat higher than the alarm value, and a charging/letdown mis-match of 180 gpm is arbitrarily chosen.

Table A1-2. CVCS Malfunction Notional Plant Response with no Operator Action

	Letdown isolation		Controlling pressure LT fails low	
Event	Time estimate (minutes)	Notes	Time estimate (minutes)	Notes
PRZ Hi-level Alarm	10	Assumed charging / letdown mis-match = 120 gpm	7	Assumed charging / letdown mis-match = 180 gpm; based on a look at the available material, it does not appear that the failed level transmitter will affect reactor trip on high-level
Reactor trip	34		23	
PRZ overfill	80		53	

Appendix 2. Human Error Probability Estimates

5 new human failure events (HFE) are introduced into the Byron SPAR model due to the new event tree models, namely ISINJ, SLBIC and SLBOC. The human error probabilities of these events are calculated in this appendix and are used in the model.

SAPHIRE software has a convenient tool to calculate and document basic event probabilities for HEPs, using the basic principles of the SPAR-H model (Reference 5). This tool is used to calculate the 5 HEPs and the calculation details are presented in Tables A2-1 through A2-5.

In addition, an exhaustive HRA analysis of these HFEs has been performed using the details of the procedure steps, various HRA quantification models, and the similar events that either happened or simulated in the past. This work is documented in a report in ADAMS as ML16223A729. The observations and insights from this reference are used in making the calculations in this appendix. The reference also provides alternative calculations for the HEPs, using different sets of plausible assumptions. This provides a good insight into the modeling uncertainties inherent in the HEP estimations in PRA.

The HEPs calculated in Tables A2-1 through A2-5 are summarized below.

OPA-1	SI termination for ISINJ	1.00E-03
OPA-2	SI termination for SLBOC	3.73E-03
OPA-3	Open Block Valves	2.25E-03
OPA-4	SI termination for ISINJ - dependent on failure of OPA-3	5.01E-01
OPA-5	SI termination for SLBIC	1.25E-03

Table A2-1. Estimation of OPA-1 HEP

Description/ Shaping Factor	Distrib. Type/ PSF	Probability/ %	Multiplier	Event Notes
SI termination for ISINJ	Log Normal	1.00E-03		
Diagnosis is Modeled		Nominal Value	1.00E-02	Time available for both OPA-3 and OPA-1 is 40 minutes.
Available Time	Extra time	100%	1.00E-01	The justification for extra time is that based on the KAERI's time data that calculated an estimated about 306 seconds of time required. The KAERI's time is considered as optimistic for real events. A factor of 3 is multiplied to the 306 seconds to cover the time distribution. The time available is 40 minutes. This results in extra time.
Stress/Stressors	Nominal	100%	1.00E+00	
Complexity	Nominal	100%	1.00E+00	
Experience/Training	Nominal	100%	1.00E+00	
Procedures	Diagnostic/Symptom oriented	100%	5.00E-01	
Ergonomics/HMI	Nominal	100%	1.00E+00	
Fitness for Duty	Nominal	100%	1.00E+00	
Work Processes	Nominal	100%	1.00E+00	
Action is Modeled		Nominal Value	1.00E-03	
Available Time	Nominal time	100%	1.00E+00	
Stress/Stressors	Nominal	100%	1.00E+00	
Complexity	Nominal	100%	1.00E+00	
Experience/Training	High	100%	5.00E-01	Terminate SI is frequently trained in simulator training.
Procedures	Diagnostic/Symptom oriented	100%	1.00E+00	
Ergonomics/HMI	Nominal	100%	1.00E+00	
Fitness for Duty	Nominal	100%	1.00E+00	
Work Processes	Nominal	100%	1.00E+00	

Table A2-2. Estimation of OPA-2 HEP

Description/ Shaping Factor	Distrib. Type/ PSF	Probability/ %	Multiplier	Event Notes
SI termination for SLBOC	Log Normal	3.73E-03		
Diagnosis is Modeled		Nominal Value	1.00E-02	5% probability that false secondary radiation alarms are create due to steam misleading operators. This is reflected in the model as Ergonomics/HMI as missing/misleading.
Available Time	Nominal time	5%	1.00E+00	This is to account for the situation that a MSLB outside of containment may cause the secondary radiation monitors to fail high due to elevated area temperatures. The false radiation indications affect the diagnostic step on transitioning from E-2 to E-3. The false alarms will slow down the operator in implementing E-2.
Available Time	Extra time	95%	1.00E-01	The extra time is based on the Turkey Point 3 event that takes 15 minutes to isolate the SI. The 15 minutes is considered equivalent to the time spent on E-2. The 15 minutes is considered not covering the time required to implement the E-0 and ES-1.1. Based on the 40 minutes time available, the Extra Time is determined as the status.
Stress/Stressors	High	5%	2.00E+00	This is to account for the stress based on the perception of encountering a MSLB & SGTR event.
Stress/Stressors	Nominal	95%	1.00E+00	
Complexity	Nominal	100%	1.00E+00	
Experience/Training	Nominal	100%	1.00E+00	
Procedures	Diagnostic/Symptom oriented	100%	5.00E-01	
Ergonomics/HMI	Missing/misleading	5%	5.00E+01	This is to account for the situation that a MSLB outside of containment may cause the secondary radiation monitors to fail high due to elevated area temperatures. The false radiation indications affect the diagnostic step on transitioning from E-2 to E-3.
Ergonomics/HMI	Nominal	95%	1.00E+00	
Fitness for Duty	Nominal	100%	1.00E+00	
Work Processes	Nominal	100%	1.00E+00	
Action is Modeled		Nominal Value	1.00E-03	

Available Time	Nominal time	100%	1.00E+00	
Stress/Stressors	High	5%	2.00E+00	
Stress/Stressors	Nominal	95%	1.00E+00	
Complexity	Nominal	100%	1.00E+00	
Experience/Training	High	100%	5.00E-01	Terminate SI is a frequently trained action in simulator training.
Procedures	Diagnostic/Symptom oriented	100%	1.00E+00	
Ergonomics/HMI	Nominal	100%	1.00E+00	
Fitness for Duty	Nominal	100%	1.00E+00	
Work Processes	Nominal	100%	1.00E+00	

Table A2-3. Estimation of OPA-3 HEP

Description/ Shaping Factor	Distrib. Type/ PSF	Probability/ %	Multiplier	Event Notes
Open Block Valves	Log Normal	2.25E-03		
Diagnosis is Modeled		Nominal Value	1.00E-02	
Available Time	Nominal time	100%	1.00E+00	The time available is 20 minutes. The time required is based on the 3 times of KAREI data that is about 480 seconds (6 minutes). This results in a Normal time available.
Stress/Stressors	Nominal	100%	1.00E+00	
Complexity	Obvious diagnosis	100%	1.00E-01	This is a procedure led detection on PZR block valve status indication.
Experience/Training	High	100%	5.00E-01	*This step is in E-0 that almost all licensed operator* simulator training scenario will run through this step.
Procedures	Diagnostic/Symptom oriented	100%	5.00E-01	
Ergonomics/HMI	Nominal	100%	1.00E+00	
Fitness for Duty	Nominal	100%	1.00E+00	
Work Processes	Nominal	100%	1.00E+00	
Action is Modeled		Nominal Value	1.00E-03	
Available Time	Nominal time	100%	1.00E+00	
Stress/Stressors	Nominal	100%	1.00E+00	
Complexity	Nominal	100%	1.00E+00	
Experience/Training	Nominal	100%	1.00E+00	
Procedures	Diagnostic/Symptom oriented	100%	1.00E+00	
Ergonomics/HMI	Nominal	100%	1.00E+00	
Fitness for Duty	Nominal	100%	1.00E+00	
Work Processes	Nominal	100%	1.00E+00	
Dependency is not Modeled				

Table A2-4. Estimation of OPA-4 HEP

Description/ Shaping Factor	Distrib. Type/ PSF	Probability/ %	Multiplier	Event Notes
SI termination for ISINJ - dependent on failure of OPA-3	Log Normal	5.01E-01		
Diagnosis is Modeled		Nominal Value	1.00E-02	
Available Time	Extra time	100%	1.00E-01	The justification for extra time is that based on the KAERI's time data that calculated an estimated about 306 seconds of time required. The KAERI's time is considered as optimistic for real events. A factor of 3 is multiplied to the 306 seconds to cover the time distribution. The time available is 40 minutes. This results in extra time.
Stress/Stressors	Nominal	100%	1.00E+00	
Complexity	Nominal	100%	1.00E+00	
Experience/Training	Nominal	100%	1.00E+00	
Procedures	Diagnostic/Symptom oriented	100%	5.00E-01	
Ergonomics/HMI	Nominal	100%	1.00E+00	
Fitness for Duty	Nominal	100%	1.00E+00	
Work Processes	Nominal	100%	1.00E+00	
Action is Modeled		Nominal Value	1.00E-03	
Available Time	Nominal time	100%	1.00E+00	
Stress/Stressors	Nominal	100%	1.00E+00	
Complexity	Nominal	100%	1.00E+00	
Experience/Training	High	100%	5.00E-01	Terminate SI is frequently trained in simulator training.
Procedures	Diagnostic/Symptom oriented	100%	1.00E+00	
Ergonomics/HMI	Nominal	100%	1.00E+00	
Fitness for Duty	Nominal	100%	1.00E+00	
Work Processes	Nominal	100%	1.00E+00	
Dependency is Modeled		Same Crew, Close in Time, Different Location, Additional Cues		High Dependence Dep. = (1+P)/2

Table A2-5. Estimation of OPA-5 HEP

Description/ Shaping Factor	Distrib. Type/ PSF	Probability/ %	Multiplier	Event Notes
SI termination for SLBIC	Log Normal	1.25E-03		
Diagnosis is Modeled		Nominal Value	1.00E-02	
Available Time	Extra time	100%	1.00E-01	The extra time is based on the Turkey Point 3 event that takes 15 minutes to isolate the SI. The 15 minutes is considered equivalent to the time spent on E-2. The 15 minutes is considered not covering the time required to implement the E-0 and ES-1.1. Based on the 40 minutes time available, the Extra Time is determined as the status.
Stress/Stressors	Nominal	100%	1.00E+00	
Complexity	Nominal	100%	1.00E+00	
Experience/Training	High	100%	5.00E-01	Operator runs about 3 MSLB simulator scenarios per year.
Procedures	Diagnostic/Symptom oriented	100%	5.00E-01	
Ergonomics/HMI	Nominal	100%	1.00E+00	
Fitness for Duty	Nominal	100%	1.00E+00	
Work Processes	Nominal	100%	1.00E+00	
Action is Modeled		Nominal Value	1.00E-03	
Available Time	Nominal time	100%	1.00E+00	
Stress/Stressors	Nominal	100%	1.00E+00	
Complexity	Nominal	100%	1.00E+00	
Experience/Training	High	100%	5.00E-01	Terminate SI action is frequently trained in simulator training.
Procedures	Nominal	100%	1.00E+00	
Procedures	Diagnostic/Symptom oriented	100%	1.00E+00	
Ergonomics/HMI	Nominal	100%	1.00E+00	
Fitness for Duty	Nominal	100%	1.00E+00	
Work Processes	Nominal	100%	1.00E+00	

Appendix 3. Classification of Plant Conditions from UFSAR

This information is taken from the Byron-Braidwood UFSAR.

CHAPTER 15.0 - ACCIDENT ANALYSES

15.1.1 Classification of Plant Conditions

Since 1970, the American Nuclear Society (ANS) classification of plant conditions has been used which divides plant conditions into four categories in accordance with anticipated frequency of occurrence and potential radiological consequences to the public. The four categories are as follows:

Condition I:	Normal Operation and Operational Transients.
Condition II:	Faults of Moderate Frequency.
Condition III:	Infrequent Faults.
Condition IV:	Limiting Faults.

The basic principle applied in relating design requirements to each of the conditions is that the most probable occurrences should yield the least radiological risk to the public and those extreme situations having the potential for the greatest risk to the public shall be those least likely to occur. Where applicable, reactor trip system and engineered safeguards functioning is assumed to the extent allowed by considerations such as the single failure criterion, in fulfilling this principle.

15.1.1.1 Condition I - Normal Operation and Operational Transients

Condition I occurrences are those which are expected frequently or regularly in the course of power operation, refueling, maintenance, or maneuvering of the plant. As such, Condition I occurrences are accommodated with margin between any plant parameter and the value of that parameter which would require either automatic or manual protective action. Inasmuch as Condition I occurrences occur frequently or regularly, they must be considered from the point of view of affecting the consequences of fault conditions (Conditions II, III and IV). In this regard, analysis of each fault condition described is generally based on a conservative set of initial conditions corresponding to adverse conditions which can occur during Condition I operation.

A typical list of Condition I events is listed below:

a. Steady-state and shutdown operations

   1. Power operation (>5 to 100% of rated thermal power),
   2. Startup (k_{eff} > 0.99 to <=5% of rated thermal power),
   3. Hot standby (subcritical, residual heat removal system isolated),
   4. Hot shutdown (subcritical, residual heat removal system in operation),

5. Cold shutdown (subcritical, residual heat removal system in operation), and
6. Refueling

b. Operation with permissible deviations

Various deviations which may occur during continued operation as permitted by the plant Technical Specifications must be considered in conjunction with other operational modes. These include:

1. Operation with components or systems out of service,
2. Radioactivity in the reactor coolant, due to leakage from fuel with cladding defects,
   (a) Fission products
   (b) Corrosion products
   (c) Tritium
3. Operation with steam generator leaks up to the maximum allowed by the Technical Specification 3.4.13, and
4. Testing as allowed by Technical Specifications and the Technical Requirements Manual (TRM)

c. Operational transients

1. Plant heatup and cooldown (up to 100degF/hour for the reactor coolant system; 200 degF/hour for the pressurizer during cooldown and 100 degF/hour for the pressurizer during heatup),
2. Step load changes (up to ± 10%),
3. Ramp load changes (up to 5%/minute), and
4. Load rejection up to and including design full load rejection transient

15.1.1.2 Condition II - Faults of Moderate Frequency

These faults, at worst, result in the reactor trip with the plant being capable of returning to operation. By definition, these faults (or events) do not propagate to cause a more serious fault, i.e., Condition III or IV events. In addition, Condition II events are not expected to result in fuel rod failures or reactor coolant system or secondary system overpressurization.

For the purposes of this report, the following faults are included in this category:

a. Feedwater system malfunctions that result in a decrease in feedwater temperature (Subsection 15.1.1),
b. Feedwater system malfunctions that result in an increase in feedwater flow (Subsection 15.1.2),
c. Excessive increase in secondary steam flow (Subsection 15.1.3),

d. Inadvertent opening of a steam generator relief or safety valve (Subsection 15.1.4),
e. Loss of external electrical load (Subsection 15.2.2),
f. Turbine trip (Subsection 15.2.3),
g. Inadvertent closure of main steam isolation valves (Subsection 15.2.4),
h. Loss of condenser vacuum and other events resulting in turbine trip (Subsection 15.2.5),
i. Loss of nonemergency ac power to the station auxiliaries (Subsection 15.2.6),
j. Loss of normal feedwater flow (Subsection 15.2.7),
k. Partial loss of forced reactor coolant flow (Subsection 15.3.1),
l. Uncontrolled rod cluster control assembly bank withdrawal at a subcritical or low power startup condition (Subsection 15.4.1),
m. Uncontrolled rod cluster control assembly bank withdrawal at power (Subsection 15.4.2),
n. Rod cluster control assembly misalignment (dropped full length assembly, dropped full length assembly bank, or statically misaligned full length assembly) (Subsection 15.4.3),
o. Deleted
p. Chemical and volume control system malfunction that results in a decrease in the boron concentration in the reactor coolant (Subsection 15.4.6),
q. Inadvertent operation of the emergency core cooling system during power operation (Subsection 15.5.1),
r. Chemical and volume control system malfunction that increases reactor coolant inventory (Subsection 15.5.2),
s. Inadvertent opening of a pressurizer safety or relief valve (Section 15.6.1), and
t. Break in instrument line or other lines from reactor coolant pressure boundary that penetrate containment (Subsection 15.6.2).

15.1.1.3 Condition III - Infrequent Faults

By definition Condition III occurrences are faults which may occur very infrequently during the life of the plant. They will be accommodated with the failure of only a small fraction of the fuel rods although sufficient fuel damage might occur to preclude resumption of the operation for a considerable outage time. The release of radioactivity will not be sufficient to interrupt or restrict public use of those areas beyond the exclusion radius.

A Condition III fault will not, by itself, generate a Condition IV fault or result in consequential loss of function of the reactor coolant system or containment barriers. For the purposes of this report the following faults are included in this category:

a. Steam system piping failure from zero power and full power (minor) (Subsections 15.1.5 and 15.1.6),
b. Complete loss of forced reactor coolant flow (Subsection 15.3.2),
c. Rod cluster control assembly misalignment (single rod cluster control assembly withdrawal at full power) (Subsection 15.4.3),

d. Inadvertent loading and operation of a fuel assembly in an improper position (Subsection 15.4.7),
e. Loss of coolant accidents resulting from a spectrum of postulated piping breaks within the reactor coolant pressure boundary (small break) (Subsection 15.6.5),
f. Gaseous radwaste system leak or failure (Subsection 15.7.1),
g. Liquid radwaste system leak or failure (Subsection 15.7.2),
h. Postulated radioactive releases due to liquid tank failures (Subsection 15.7.3), and
i. Spent fuel cask drop accidents (Subsection 15.7.5).

15.1.1.4 Condition IV - Limiting Faults

Condition IV occurrences are faults which are not expected to take place, but are postulated because their consequences would include the potential of the release of significant amounts of radioactive material. They are the most drastic which must be designed against and represent limiting design cases.

Condition IV faults are not to cause a fission product release to the environment resulting in an undue risk to public health and safety in excess of guidelines values of 10 CFR 100 for TID-14844 based dose analyses and 10 CFR 50.67 for AST based analyses.

A single Condition IV fault is not to cause a consequential loss of required functions of systems needed to cope with the fault including those of the emergency core cooling system and the containment.

For the purposes of this report, the following faults have been classified in this category:

a) Steam system piping failure from zero power and full power (major) (Subsections 15.1.5 and 15.1.6),
b) Feedwater system pipe break (Subsection 15.2.8),
c) Reactor coolant pump shaft seizure (locked rotor) (Subsection 15.3.3),
d) Reactor coolant pump shaft break (Subsection 15.3.4),
e) Spectrum of rod cluster control assembly ejection accidents (Subsection 15.4.8),
f) Steam generator tube failure (Subsection 15.6.3),
g) Loss of coolant accidents resulting from the spectrum of postulated piping breaks within the reactor coolant pressure boundary (large break) (Subsection 15.6.5), and
h) Design basis fuel handling accidents (Subsection 15.7.4).

Appendix 4. Byron SPAR Model

The existing Byron Standardized Plant Analysis Risk (SPAR) model (Version 8.27) was supplemented with three additional initiating events for completeness related to this analysis, by considering those events that are of relevance to this analysis that are typically subsumed in the General Transients category or not modeled in SPAR models. These three initiating events are inadvertent safety injection (SI), steam line breaks (SLB) inside and outside the containment.

In steam generator tube rupture (SGTR), SLB, and spurious SI event tree models, an event tree node for failure of SI termination exists. For SGTR this node determines what downstream success paths are viable (residual heat removal versus RWST refill). For SLB and spurious SI, if this node fails, the event is assumed to continue as a SLOCA, with the underlying assumption that the SRVs will pass water and continue to do so. Note that this failure is separate from SLOCA through a failed-open PORV path. Section 2.2.1 summarizes the success criteria used for potential pressurizer overfill and SI termination.

D.1 Inadvertent Safety Injection Event Tree

The following provides a description of the SPAR model Inadvertent Safety Injection event tree. Event tree specific success criteria are provided, followed by a description of the event tree headings and the event tree structure. Figure D-1 shows the inadvertent safety injection event tree.

D.1.1 Success Criteria

Successful plant response to an inadvertent safety injection initiating event requires successful reactivity control, early decay heat removal and inventory control, and long-term decay heat removal. If the reactor fails to trip and insert enough negative reactivity by the control rods to shut down the reactor, the sequence transfers to the ATWS event tree. Successful operation of secondary cooling (AFW) provides initial decay heat removal. Control of makeup involves several operator actions and/or equipment operations. Unblocking PORV block valves (UNBLOCK), if closed, allows PORVs to relieve pressure buildup in lieu of opening the code safeties. If either PORV path is available and functional, opening of the code safeties is assumed to be precluded.

Reduction of injection flow up to and including termination (CSI) is then required to prevent the charging/safety injection pumps from overfilling the pressurizer. Overfilling of the pressurizer can cause an excessive number of relief/safety valve lifts and potentially lead to release of water through the valves. Passing water through these valves is expected to result in an elevated failure rate. The position of the PORVs and code safety valves is then determined with unmitigated failures in the open position transferring to the small loss of coolant event tree (SLOCA) for additional evaluation.

Feed and bleed cooling can provide successful decay heat removal given secondary cooling is unavailable or the RCS is experiencing loss of inventory. For feed and bleed cooling success, a single PORV is required to open if a charging pump is available for injection. Two PORVs are required to open and remove the decay heat during small loss-of-coolant events if only the safety injection pumps are available for injection to replenish the lost RCS inventory.

D.1.2 General Description/Philosophy

D.1.3 Top Event Descriptions

The inadvertent safety injection event tree has the following events arranged in the approximate order in which they would be expected to occur following the event.

IE-ISINJ	Initiating event is an inadvertent safety injection. The primary issue associated with an inadvertent safety injection is the starting of all ECCS equipment including the CVCS and SI pumps and isolation of letdown. Continued operation of high-pressure injection pumps will lead to the pressurizer going solid and the passing of water through the PORVs and/or code safety valves.
RPS	This top represents the success or failure of the reactor protection system (RPS) to insert enough negative reactivity by the control rods to shut down the reactor.
AFW	Auxiliary feedwater (AFW) system is used to remove decay heat via the steam generators given MFW is not available. The main feedwater system will isolate given a reactor trip. This will require the use of the AFW system to provide flow to the steam generators. Success implies automatic actuation and operation of the AFW system. The AFW system supplies sufficient cooling water to the steam generators to remove decay heat from the reactor. The success criteria are one-of-two AFW trains to three-of-four steam generators.
UNBLOCK	This top represents the operator action to open, if closed, the pressurizer block valves. Success is defined as at least one PORV path available for pressure relief or depressurization.
CSI	This top event represents the success or failure of an operator to control the injection of the high pressure injection pumps. Success implies the operator took control of the high-pressure pumps (CVCS & SI) to either terminate flow or to lower the flow rate of the pumps.
FAB	Success or failure of feed and bleed cooling is represented by this top event. Feed and bleed cooling is required given secondary cooling is unavailable. Success requires one-of-two PORVs for successful depressurization when a charging pump is available for injection. Both PORVs are required for depressurization when only the SI pumps are available for injection. An operator is required to open the PORVs and the PORV block valves if they are closed. Success also requires the CVCS/SI system(s) to provide flow to the RCS cold legs.

HPI This top event represents the success or failure of the high pressure injection system to provide makeup water to the RCS. Success implies automatic actuation and operation of the HPI system (i.e., safety injection (SI) pumps and charging (CVC) pumps). The pumps take suction from the refueling water storage tank (RWST) and provide flow to the RCS cold legs. The HPI system provides sufficient water to keep the core covered. The success criteria are one-of-two SI trains or one-of-two CVC trains supplying at least two-of-four cold legs.

HPR This top event represents the success or failure of high pressure recirculation. Success requires the HPI pumps (SI or CVC pumps) to take suction from the discharge of the RHR pumps and deliver the water to the RCS. HPR will provide long-term cooling for the reactor given the HPI system was successful in supplying early makeup water to the reactor. HPR is required if residual heat removal cannot be established. The decay heat will be removed from the containment sump by the RHR pump train heat exchangers. An operator action is required to align the RHR pump discharge to the HPI pump suction and verify that the containment sump valves are open and the RWST suction valves are closed. The success criteria are one-of-two RHR trains (and their respective heat exchangers) providing flow to one-of-four HPI trains (one-of-two SI or one-of-two CVC trains).

D.2 Steam Line Break Containment Event Tree

The following provides a description of the SPAR model Steam Line Break event tree. Event tree specific success criteria are provided, followed by a description of the event tree headings and the event tree structure. Figure D-2 shows the steam line break event tree. **NOTE**: The event tree logic is identical between the steam line breaks inside and outside containment. The difference is contained in the associated fault tree. The initiator specific logic is turned on/off with the use of house events.

D.2.1 Success Criteria

Successful plant response to a steam line break initiating event requires successful reactivity control, early decay heat removal and inventory control, and long-term decay heat removal. If the reactor fails to trip and insert enough negative reactivity by the control rods to shut down the reactor, the sequence transfers to the ATWS event tree. Due to the rapid decrease in steam pressure associated with a steam line break; this event has the potential of inducing a steam generator tube rupture (SGTR). If a steam generator tube rupture does occur, the sequence is transferred out to the SGTR event tree for more detailed evaluation. Successful operation of secondary cooling (AFW) provides initial decay heat removal. Isolation of the faulted steam generator (MSI) is initiated to control cooldown and to terminate the mass and energy releases. The feed flow paths into as well as the steam paths exiting the faulted generator must be isolated. The steam line isolation criterion is dependent on whether the break location is inside or outside the containment. Additionally, failure to isolate the faulted steam generator following a break is assumed to require high-pressure injection (HPI) to recover pressurizer level due to the RPV coolant shrinkage. Finally, an unisolated break inside the containment is also assumed to actuate containment sprays and lead to the need for sump recirculation (HPR).

The status of the reactor pressure vessel (RPV) coolant inventory is evaluated in the C-SLOCA node. Primary coolant shrinkage due to the rapid cooldown is expected to result in a safety injection signal on low pressurizer level. If this injection is not controlled or terminated, overfilling of the pressurizer will occur. Opening and reclosing of PORVs to relieve pressure is also considered within this node. If RPV inventory is being lost, the scenario transfers to the SLOCA event tree for further evaluation.

Feed and bleed cooling can provide successful decay heat removal given secondary cooling is unavailable or the RCS is experiencing loss of inventory. For feed and bleed cooling success, a single PORV is required to open if a charging pump is available for injection. Two PORVs are required to open and remove the decay heat during small loss-of-coolant events if only the safety injection pumps are available for injection to replenish the lost RCS inventory.

D.2.2 General Description/Philosophy

D.2.3 Top Event Descriptions

The steam line break event tree has the following events arranged in the approximate order in which they would be expected to occur following the event.

IE-SLBIC	Initiating event steam line break inside/outside containment. (IE-SLBIC/IE-SLBOC). The steam line break event tree structures are nearly identical. The only differences are in the MSI and HPR fault tree logic. Logic differences within the MSI fault tree are actuated using initiator specific house events. The HPR fault tree is set to FALSE in the SLBOC event tree.
RPS	This top represents the success or failure of the reactor protection system (RPS) to insert enough negative reactivity by the control rods to shut down the reactor.
ISGTR	This top represents the likelihood of the event causing a steam generator tube rupture. The increased differential pressure across the steam generator tube sheet/tubes from the initiator increases the likelihood of a tube rupture.
AFW	Auxiliary feedwater (AFW) system is used to remove decay heat via the steam generators given MFW is not available. The main feedwater system will isolate given a reactor trip. This will require the use of the AFW system to provide flow to the steam generators. Success implies automatic actuation and operation of the AFW system. The AFW system supplies sufficient cooling water to the steam generators to remove decay heat from the reactor. The success criteria are one-of-two AFW trains to three-of-four steam generators.
MSI	This top represents the operator action to isolate the steam line that contains the break. This top also contains the hardware required for isolation. For steam line breaks outside containment (SLBOC), all MSIVs must close to isolate steam

flow. For steam line breaks inside containment (SLBIC), only the MSIV of the faulted steam generator needs to be closed. In addition, feedwater flow to the faulted steam generator must also be secured.

C-SLOCA This top event represents the various pathways leading to a consequential small loss of coolant (SLOCA) event. These paths include failure of operators to control the injection of the high pressure injection pumps. Success implies operators take control of the HPI pumps and lower the flow rate of the pumps. Failure of the pressurizer PORVs to reclose if opened also lead to a SLOCA as well as a loss reactor coolant pump seal cooling.

FAB Success or failure of feed and bleed cooling is represented by this top event. Feed and bleed cooling is required given secondary cooling is unavailable. Success requires one-of-two PORVs for successful depressurization when a charging pump is available for injection. Both PORVs are required for depressurization when only the SI pumps are available for injection. An operator is required to open the PORVs and the PORV block valves if they are closed. Success also requires the CVCS/SI system(s) to provide flow to the RCS cold legs.

HPI This top event represents the success or failure of the high pressure injection system to provide makeup water to the RCS. Success implies automatic actuation and operation of the HPI system (i.e., safety injection (SI) pumps and charging (CVC) pumps). The pumps take suction from the refueling water storage tank (RWST) and provide flow to the RCS cold legs. The HPI system provides sufficient water to keep the core covered. The success criteria are one-of-two SI trains or one-of-two CVC trains supplying at least two-of-four cold legs.

HPR This top event represents the success or failure of high pressure recirculation. Success requires the HPI pumps (SI or CVC pumps) to take suction from the discharge of the RHR pumps and deliver the water to the RCS. HPR will provide long-term cooling for the reactor given the HPI system was successful in supplying early makeup water to the reactor. HPR is required if residual heat removal cannot be established. The decay heat will be removed from the containment sump by the RHR pump train heat exchangers. An operator action is required to align the RHR pump discharge to the HPI pump suction and verify that the containment sump valves are open and the RWST suction valves are closed. The success criteria are one-of-two RHR trains (and their respective heat exchangers) providing flow to one-of-four HPI trains (one-of-two SI or one-of-two CVC trains).

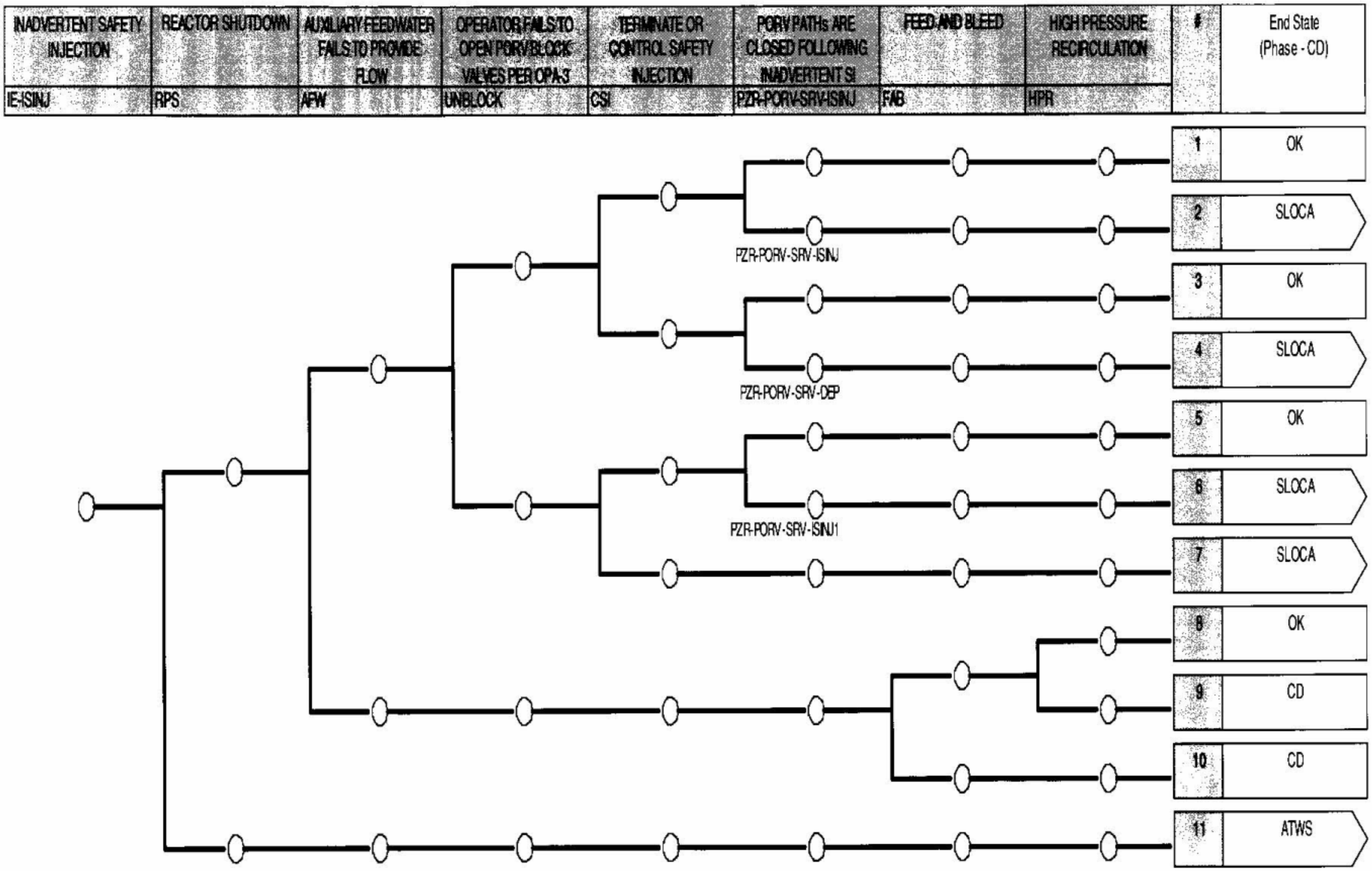

Figure D-1: Inadvertent Safety Injection Event Tree

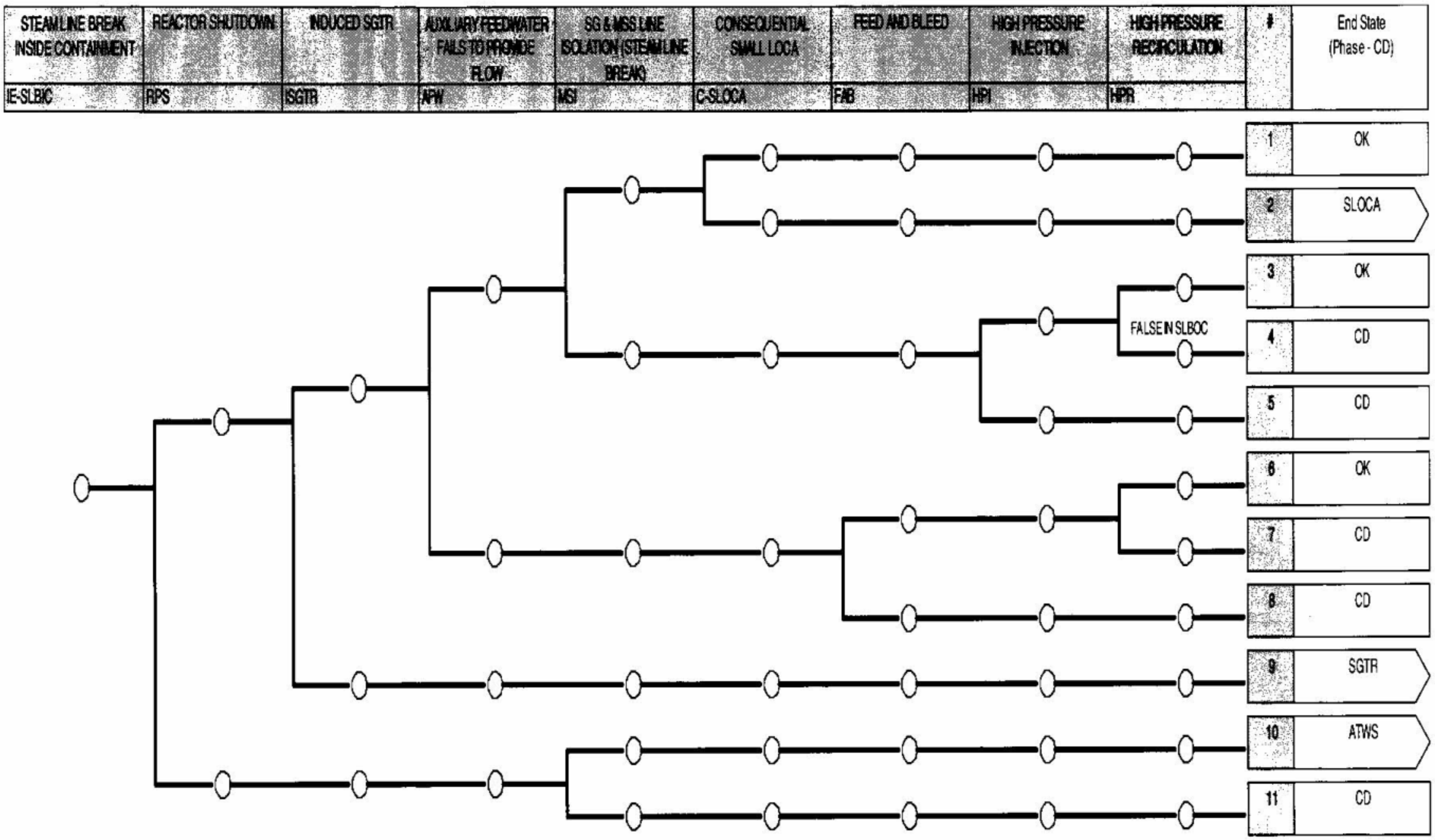

Figure D-2: Steam Line Break Event Tree

D.3 Notes on SRV Failures

The small LOCA frequency we use is for pipe-break events, and we currently have zero events cataloged in our initiating event database (either initial plant fault or functional impact)

INL specifically tracks the stuck open valve cases and have no initial plant fault initiator events for these cases. We do have some functional impact events coded, giving the following

PWR, G2, Stuck open safety valve, frequency 1.5E-3 per rcry

PWR, G4 Stuck open pressurizer PORV frequency 9.0E-4 rcry

The functional impact category should provide an occurrence bound on the idea of consequential small LOCAs from the TRANS/stuck open RV sequences which are included in the model.

There are 2 PWR stuck open SV events as follows, by initial plant fault:

1) Calvert Cliffs 1, 1994, QR5 turbine trip, actually described as the SV leaking by its seat at 25 gpm, so not really much of a stuck open SV event
2) Fort Calhoun, 1992, QC4 loss of ac I&C bus, closer to a real SV event, with failure to re-seat and a leak rate of 200 gpm.

So a conditional probability could be worked out using either or both of these events. The question would be what to use in the denominator. Could use the number of Q events, or some sub-set. Would lead to a higher conditional probability than we now calculate.

For PWR stuck open PORV we have the following event:

1) Calvert Cliffs, 2006, QR0 RCS high pressure (RPS trip), the PORV remained open for about 90 seconds, allowing pressure to drop to 1500 psia, when it should have re-closed at about 2400 psia. So this event was really only a temporary stuck open PORV.

Again, what to use in the denominator? All Q events?

So maybe nothing more needs to be done, but the above data could be used if needed.

D.4 Notes on How Model Cases Were Run

The original Byron SPAR model version 8.27 already has 2 basic events for pressurizer PROV block valves being in a closed position during power operation. These basic events are assigned the probability of 0.0633 in the modified SPAR model. However, these probabilities are not used in calculation of the CDF for the 3 plant configuration cases Case-0, Case-1 and Case-2. For these cases, three change sets are used to replace these basic events with combinations of TRUE and FALSE values (corresponding to failure probabilities of 1 and 0, respectively). Table D.4-1 shows the assigned values discussed above.

Table D.4-1 Basic Event Probabilities Assigned to Block Valves via Change Sets for Cases 0, 1 and 2

		Basic Event Probability			
Basic Event Name	**Description**	**Base SPAR Model**	**Case-0**	**Case-1**	**Case-2**
PPR-MOV-FC-8000A	PORV 8000A BLOCK VALVE CLOSED DURING POWER	6.33E-02	FALSE	TRUE	TRUE
PPR-MOV-FC-8000B	PORV 8000B BLOCK VALVE CLOSED DURING POWER	6.33E-02	FALSE	FALSE	TRUE

These cases are solved with a cutoff probability of 1E-12 using the SAPHIRE software.

For Cases B and C, additional change sets are made to replace the HEP values with new ones. These replacements are shown in Table 4-2.

Thus, a total of 9 runs are made with SAPHIRE version 8.1.4.4, three plant configuration cases for each of the Cases A, B and C which are defined in Section 4. The resulting CDFs are obtained from the minimal cutsets as given in the standard output Table J-2 of SAPHIRE.

The dependent HFE event, OPA-4, was placed in the cutsets through a rule in the event tree post-processing file.

LAST PAGE

From:	West, Steven
Sent:	Wednesday, August 17, 2016 10:13 AM
To:	Clark, Theresa
Cc:	Holahan, Gary; Scarbrough, Thomas; Spencer, Michael; West, Steven
Subject:	Panel cover letter - suggested edits
Attachments:	cover memo (MASTER) WEST 2016 08 17.docx

August XX, 2016

MEMORANDUM TO: Victor M. McCree
Executive Director for Operations

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

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

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

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

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

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

Commented [MAS1]: This is the title in the charter.

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

Commented [SW2]: Add ADAMS ML numbers for all referenced documents?

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

Commented [SW3]: Add titles for Brad and Tony?

CONTACT: Gary M. Holahan, OEDO

301-415-17XX

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

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

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

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

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

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

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

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

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

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

Commented [SW4]: Moved to the new sentence above. (I could go either way, but like closing the loop on the Exelon/NEI offer.)

Commented [SW5]: Include a reference to Tony's memo somewhere?

Commented [SW6]: Should we be a bit more specific about what "this issue" is? NRR's comments on our preliminary findings indicate some lack of clarity about what the issue is.

Commented [SW7]: I don't think we're talking about the actual Generic (capital G) Issues (capital I) process, or are we? The revision leaves this as a possibility, but allows other options. See also, Michael's comment below.

Commented [MAS8]: It seems appropriate to answer the 5 questions in the cover memo. Right now, they begin on page 11 of the report.

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

Commented [SW9]: In discussion with Brad Fewell at the UWC, he reiterated Exelon's position that there was no mistake, rather the staff has "reinterpreted" its positions.

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

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

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

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

Commented [SW10]: I suggest that we add the RES analysis to the second paragraph of the memo, where we describe the scope of what we reviewed, and make the conforming editorial change noted here.

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

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

Commented [MAS11]: This could be taken to mean "Generic Issues" process.

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

Commented [ST12]: We should indicate that the report includes lessons from our review.

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

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

[Patti to add concurrence page]

From: Garmoe, Alex
Sent: Wednesday, July 06, 2016 4:51 PM
To: Clark, Theresa
Cc: Holahan, Gary
Subject: RE: Question: Backfit Appeal to EDO

Theresa,

I returned Dave's call and provided your name as a point of contact should he have any questions or concerns about the backfit appeal. For your information, Dave Gullott can be reached at 630-657-2807.

Alex

From: Clark, Theresa
Sent: Wednesday, July 06, 2016 1:21 PM
To: Garmoe, Alex <Alex.Garmoe@nrc.gov>
Cc: Holahan, Gary <Gary.Holahan@nrc.gov>
Subject: RE: Question: Backfit Appeal to EDO

The short answer is no, not right now. (That's what we told Brad Fewell in a call with Gary Holahan the other day.) Todd Keene is the appointed PM from NRR, but not much has been requested him (just CAC, public meeting setup if needed). You could point Dave to me or Gary if he has any specific questions. Thanks!

From: Garmoe, Alex
Sent: Wednesday, July 06, 2016 1:16 PM
To: Clark, Theresa <Theresa.Clark@nrc.gov>
Subject: Question: Backfit Appeal to EDO

Theresa,

I received a voicemail from Dave Gullott with Exelon asking if we needed any additional info or had any questions about the backfit appeal. Since I haven't been requested to provide PM support for this appeal, I'd like to provide him with an alternate point of contact. Could you let me know who that would be so I can direct Dave their way?

Thanks,

Alexander D. Garmoe
Senior Project Manager
Generic Communications Branch (PGCB)
Division of Policy and Rulemaking (DPR)
Office of Nuclear Reactor Regulation (NRR)
Alex.Garmoe@nrc.gov | 301-415-3814

From: Roberts, Ashley
Sent: Friday, September 30, 2016 4:35 PM
To: Clark, Theresa
Cc: Valentine, Nicholee; Wiebe, Joel; Keene, Todd; Garmoe, Alex; Burnell, Scott; Abraham, Susan; Stuchell, Sheldon
Subject: RESPONSE: ACTION: pre-request - cost data for Exelon backfit

Theresa,
Below is the hours information you requested, should you need it. As you noted below, this does not include any management hours, OGC hours, or hours for any staff that did not charge to the specific CACs (like yourself as you mentioned as an ETA).

Backfit preparation for Braidwood and Byron – 1013.6 hours
Review of appeal – 202.2 hours
EDO review of appeal – 370.7 hours

Please let us know if you have questions.
Ashley

Ashley B. Roberts (Bettis)
Chief, Financial, Human Capital & Analysis Support Branch
Program Management, Policy, Development, & Analysis
Office of Nuclear Reactor Regulation
Mailstop: O13-H16M
301-415-1567

From: Abraham, Susan
Sent: Friday, September 16, 2016 9:51 AM
To: Gavrilas, Mirela <Mirela.Gavrilas@nrc.gov>; Lund, Louise <Louise.Lund@nrc.gov>; Boland, Anne <Anne.Boland@nrc.gov>; Evans, Michele <Michele.Evans@nrc.gov>
Cc: Roberts, Ashley <Ashley.RobertsBettis@nrc.gov>; Valentine, Nicholee <Nicholee.Valentine@nrc.gov>
Subject: FW: pre-request - cost data for Exelon backfit

For awareness, Susan

From: Clark, Theresa
Sent: Friday, September 16, 2016 8:58 AM
To: Roberts, Ashley <Ashley.RobertsBettis@nrc.gov>; Valentine, Nicholee <Nicholee.Valentine@nrc.gov>
Cc: Wiebe, Joel <Joel.Wiebe@nrc.gov>; Keene, Todd <Todd.Keene@nrc.gov>; Garmoe, Alex <Alex.Garmoe@nrc.gov>; Burnell, Scott <Scott.Burnell@nrc.gov>; Abraham, Susan <Susan.Abraham@nrc.gov>
Subject: pre-request - cost data for Exelon backfit

Ashley/Nikki,

We're working with OPA on some public communications regarding the Exelon backfit appeal decision by the EDO this week. They were hoping to have in their back pocket any information we had on the cost of the agency's activities related to this backfit, the NRR appeal, and the EDO appeal, as well as if they were fee billable.

I think the following are the relevant CACs, though the PMs may know better. I recognize that it will not capture everyone's hours, as managers (and I!) used different CACs, but it should include most staff time.

- **MF3206/7/8/9,** Backfit – licensing basis relis upon relief of water through the pressurizer safety valves for mitigation of…
- **MF7231/2/3/4,** Review of Appeal of Imposition of Backfit Regarding a Condition II Event that Could Cause a More Serious Event (non fee billable)
- **MF8035**, EDO Review of Appeal of Imposition of Backfit Regarding a Condition II Event that Could Cause a More Serious Event

I don't think there is a huge rush to get the information but if you could get started pulling it together that would probably make life easier in the future.

Thanks so much!

--
Theresa Valentine Clark
Executive Technical Assistant (Reactors)
U.S. Nuclear Regulatory Commission
Theresa.Clark@nrc.gov | 301-415-4048 | O-16E22

From:	Clark, Theresa
Sent:	Tuesday, September 27, 2016 12:52 PM
To:	Holian, Brian
Cc:	Inverso, Tara; West, Steven
Subject:	RE: REQUEST: brief look @ Vic msgs for NSIAC
Attachments:	VMM - NSIAC discussion key messages 10-12-16.docx

Brian, thanks again. I added some bullets to the top of p.4 on cyber. I believe they are all non-sensitive and should be the needed background. Thanks!

From: Holian, Brian
Sent: Tuesday, September 27, 2016 12:19 PM
To: West, Steven <Steven.West@nrc.gov>; Kohen, Marshall <Marshall.Kohen@nrc.gov>
Cc: Clark, Theresa <Theresa.Clark@nrc.gov>; Inverso, Tara <Tara.Inverso@nrc.gov>
Subject: Fwd: REQUEST: brief look @ Vic msgs for NSIAC

Looked ok to me on a brief scan. But needs something on cyber

I thought I responded to Vic's email and have a couple items. Tara or Steve. Place ensure Theresa has (pretty sure I put a cyber piece in)

From: "Clark, Theresa" <Theresa.Clark@nrc.gov>
Subject: REQUEST: brief look @ Vic msgs for NSIAC
Date: 27 September 2016 10:50
To: "Dean, Bill" <Bill.Dean@nrc.gov>, "Holian, Brian" <Brian.Holian@nrc.gov>, "Uhle, Jennifer" <Jennifer.Uhle@nrc.gov>
Cc: "Inverso, Tara" <Tara.Inverso@nrc.gov>

Hi there! Based on Vic's email below and your responses, I pulled together a few key messages and some background for him that could be used in his remarks on October 12 at the NSIAC event. Would you mind giving it a quick scan to see if I missed anything important from your areas of responsibility? I have not sent this to Vic yet, so he may want more/less in some areas. I committed to send it by email sometime this week, so your feedback over the next couple of days would be great. Thanks!!

Note: I don't know how long he'll have to speak, so that's why it's pretty free-form right now.

Victor M. McCree – NSIAC Remarks – October 12, 2016

KEY MESSAGES

Non-Responsive Record

-
-
-
-

BACKGROUND

Stakeholder Meeting Follow-Up

Non-Responsive Record

-
-
-

Project AIM

Non-Responsive Record

-
-
-

Byron/Braidwood Backfit Appeal Decision

- The EDO concluded that the NRC staff's position in the October 2015 backfit issued to Byron and Braidwood related to pressurizer valve performance was a new or modified interpretation of what constitutes compliance and did not provide a basis for a compliance backfit.
- This decision was communicated on 9/15/16 in publicly available letters to Exelon and NEI, as well as in a memo to the staff that also requested preparation of a plan to reevaluate the generic implications of the technical issue.
- In a separate activity, CRGR is implementing an EDO tasking to evaluate guidance, training, and knowledge management on backfitting. Multiple industry representatives participated in a 9/13/16 public meeting held by CRGR.

Subsequent License Renewal

Non-Responsive Record

Licensing Action Initiatives

Non-Responsive Record

Operator Licensing

Non-Responsive Record

Digital I&C Action Plan

Non-Responsive Record

-
-
-

50.46c Rulemaking

Non-Responsive Record

-
-

NUREG-2180 / Very Early Warning Fire Detection Systems

Non-Responsive Record

-
-

Force on Force Efficiencies/Guidance

Non-Responsive Record

-
-
-
-

Cybersecurity at Nuclear Power Plants

Non-Responsive Record

Advanced Reactor Preparations

Non-Responsive Record

New Reactor Licensing Activities

Non-Responsive Record

From: Dean, Bill
Sent: Friday, September 23, 2016 12:33 AM
To: Clark, Theresa
Subject: Fwd: ET Note on Recent Devlopments and Upcoming All Hands Meeting

First full paragraph

From: "Dean, Bill" <Bill.Dean@nrc.gov>
Subject: ET Note on Recent Devlopments and Upcoming All Hands Meeting
Date: 22 September 2016 11:22
To: "NRR Distribution" <NRRDistribution@nrc.gov>
Cc: "Johnson, Michael" <Michael.Johnson@nrc.gov>, "Uhle, Jennifer" <Jennifer.Uhle@nrc.gov>

Dear NRR Staff,

As summer is fading (today is the first day of autumn in fact), the Redskins are already 0-2 and the Nats and maybe the Orioles are dreaming of playoff baseball, we thought it was appropriate to communicate with you on a few issues.

Late last week, the EDO issued a final decision on a compliance backfit appeal filed by Exelon on a licensing basis issue at Bryon and Braidwood. He discussed his rationale for this decision in his September 16 EDO Update. It is important for you to know that the EDO's decision is based on his determination, from all the evidence in front of him, that the position taken by NRR was a "new or modified interpretation" and that we had not provided a sufficient basis to justify imposing a compliance backfit. While this is a singular case, we believe that this decision has significant implications for how we pursue our licensing and oversight role. We will have to establish clear documentation and evidence that a licensee is not in compliance with its licensing basis, which can be a challenge when dealing with plants where reasonable arguments can be made over what constitutes the licensing basis. It also marks a potential change in industry behavior, whereas in the past they have accepted certain inspection findings or licensing actions to sustain "regulatory harmony". Given the competitive nature of the current electricity markets, we can anticipate additional challenges like this occurring in the future. This case is also being looked at by the CRGR (of which Brian McDErmott is a member) which is conducting an overall evaluation of NRC's practices, training and guidance related to implementation of the backfit rule.

Non-Responsive Record

Non-Responsive Record

Non-Responsive Record

Non-Responsive Record

Non-Responsive Record

Non-Responsive Record

Non-Responsive Record

Non-Responsive Record

BILL, MICHELE, and BRIAN

From: Rihm, Roger
Sent: Thursday, September 22, 2016 1:57 PM
To: Adams, Darrell
Cc: Wu, Angela; Clark, Theresa
Subject: RE: Exelon recently appealed a compliance backfit. What is the NRC doing about that

Yes.

From: Adams, Darrell
Sent: Thursday, September 22, 2016 1:27 PM
To: Rihm, Roger <Roger.Rihm@nrc.gov>
Cc: Wu, Angela <Angela.Wu@nrc.gov>; Clark, Theresa <Theresa.Clark@nrc.gov>
Subject: Exelon recently appealed a compliance backfit. What is the NRC doing about that

Roger, per our conversation yesterday, is the last bullet accurate/sufficient?

QUESTION 43: Exelon recently appealed a compliance backfit. What is the NRC doing about that?

ANSWER:

- In October 2015, the NRC issued a compliance backfit that affected Exelon's Braidwood and Byron Stations because the NRC became aware that the accident analyses predicted water relief out of relief valves that are not qualified per ASME code to relieve water. The NRC had previously approved the analyses as part of license amendments in 2001 and 2004 under the belief that the valves were water qualified.

- Exelon exercised their right to appeal a backfit decision to the NRR Office Director. The NRR Office Director upheld the backfit based in large part on input from a backfit appeal review panel.

- Exelon then further appealed the backfit to the EDO. ~~A final decision on whether to grant the backfit appeal is expected early September.~~ **Upon appeal, the NRC determined that the agency had misapplied the compliance exception in 2015 and withdrew its application to the Braidwood and Byron plants.**

Darrell E. Adams
Senior Congressional/External Affairs Officer
U.S. Nuclear Regulatory Commission
Email: Darrell.Adams@nrc.gov
Mail Stop: 03-03E04
301-415-2339

From: Clark, Theresa
Sent: Thursday, August 11, 2016 1:04 PM
To: Sprogeris, Patricia
Cc: Holahan, Gary; West, Steven; Scarbrough, Thomas; Spencer, Michael
Subject: REQUEST: backfit appeal panel meeting w/ Vic

Patti,

Can you please arrange a meeting for the backfit appeal panel (Gary Holahan, Steve West, Tom Scarbrough, and Michael Spencer) with Vic? Mike may also wish to attend. The week of August 22 would be ideal, perhaps 11am 8/24 for half an hour if Gary and Steve can make that work (others are free). Otherwise, please work your magic to find a time. Thanks!

--

Theresa Valentine Clark
Executive Technical Assistant (Reactors)
U.S. Nuclear Regulatory Commission
Theresa.Clark@nrc.gov | 301-415-4048 | O-16E22

From: Banks, Eleasah
Sent: Friday, September 16, 2016 8:17 AM
To: RidsNrrMailCenter Resource; RidsOgcMailCenter Resource; RidsNroMailCenter Resource; RidsResPmdaMail Resource; RidsResOd Resource; RidsNmssOd Resource; RidsRgn1MailCenter Resource; RidsRgn2MailCenter Resource; RidsRgn3MailCenter Resource; RidsRgn4MailCenter Resource; RidsNrrDorlLpl3-2 Resource; RidsNrrPMByron Resource; RidsNrrPMBraidwood Resource; RidsNrrDss Resource; RidsNrrDe Resource; RidsNrrDpr Resource; RidsNrrDorl Resource; Garmoe, Alex; Keene, Todd; Gody, Tony; Gendelman, Adam; Mizuno, Beth; Correia, Richard; West, Khadijah; Bailey, Marissa; Scarbrough, Thomas; Spencer, Michael; Clark, Theresa
Subject: Appeal of Backfit Imposed in Braidwood and Byron Stations (To: William Dean, From: Victor McCree)

Date: September 15, 2016

Memorandum To: William M. Dean

From: Victor M. McCree

Subject: Appeal of Backfit Imposed in Braidwood and Byron Stations (To: William Dean, From: Victor McCree)

This document is publicly available in ADAMS

View ADAMS P8 Properties ML16246A247
Open ADAMS P8 Document (Appeal of Backfit Imposed in Braidwood and Byron Stations (To: William Dean, From: Victor McCree))

From:	Banks, Eleasah
Sent:	Friday, September 16, 2016 9:07 AM
To:	RidsNrrMailCenter Resource; RidsOgcMailCenter Resource; RidsNroMailCenter Resource; RidsResPmdaMail Resource; RidsResOd Resource; RidsNmssOd Resource; RidsRgn1MailCenter Resource; RidsRgn2MailCenter Resource; RidsRgn3MailCenter Resource; RidsRgn4MailCenter Resource; RidsNrrDorlLpl3-2 Resource; RidsNrrPMByron Resource; RidsNrrPMBraidwood Resource; RidsNrrDss Resource; RidsNrrDe Resource; RidsNrrDpr Resource; RidsNrrDorl Resource; Garmoe, Alex; Keene, Todd; Gody, Tony; Gendelman, Adam; Mizuno, Beth; Correia, Richard; West, Khadijah; Bailey, Marissa; Scarbrough, Thomas; Spencer, Michael; Clark, Theresa
Subject:	Backfit Appeal Review Panel Findings (Byron and Braidwood)

Date: September 15, 2016

Memorandum To: J. Bradley

From: Victor M. McCree

Subject: Backfit Appeal Review Panel Findings (Byron and Braidwood)

This package, consisting of 5 documents (ML16243A067, ML16236A202, ML16236A208, ML16214A199, and ML16173A311), is publicly available in ADAMS.

View ADAMS P8 Properties ML16236A198

Open ADAMS P8 Package (Backfit Appeal Review Panel Findings (Byron and Braidwood))

From: Sprogeris, Patricia
Sent: Wednesday, August 24, 2016 1:43 PM
Cc: Holahan, Gary; West, Steven; Scarbrough, Thomas; Spencer, Michael; Clark, Theresa
Subject: FW: Backfit Appeal Review Panel Findings Associated with Byron & Braidwood

I am so sorry, I forgot to put the cc's in before hitting send. The package has been dispatched per below.

Thank you, Patti

From: Sprogeris, Patricia
Sent: Wednesday, August 24, 2016 1:41 PM
To: RidsNrrOd Resource <RidsNrrOd.Resource@nrc.gov>; Correia, Richard <Richard.Correia@nrc.gov>; Mizuno, Geary <Geary.Mizuno@nrc.gov>; Lewis, Robert <Robert.Lewis@nrc.gov>; McGinty, Tim <Tim.McGinty@nrc.gov>; RidsNroOd Resource <RidsNroOd.Resource@nrc.gov>; Johnson, Michael <Michael.Johnson@nrc.gov>; Lubinski, John <John.Lubinski@nrc.gov>; Mayfield, Michael <Michael.Mayfield@nrc.gov>; Tracy, Glenn <Glenn.Tracy@nrc.gov>; RidsResOd Resource <RidsResOd.Resource@nrc.gov>; RidsOgcMailCenter Resource <RidsOgcMailCenter.Resource@nrc.gov>
Subject: Backfit Appeal Review Panel Findings Associated with Byron & Braidwood

Date: August 24, 2016

From: Gary M. Holahan
K. Steven West
Thomas G. Scarbrough
Michael A. Spencer
Theresa Valentine Clark

This package, consisting of 5 documents (ML16243A067, ML16236A202, ML16236A208, ML16214A199, and ML16173A311), is publicly available in ADAMS.

View ADAMS P8 Properties ML16236A198
Open ADAMS P8 Package (Backfit Appeal Review Panel Findings (Byron and Braidwood))

Thank you, Patti

Patti Sprogeris
Assistant to Michael R. Johnson
Office of the Executive Director for Operations
301-415-1713

From:	Banks, Eleasah
Sent:	Friday, September 16, 2016 9:03 AM
To:	RidsNrrMailCenter Resource; RidsOgcMailCenter Resource; RidsNroMailCenter Resource; RidsResPmdaMail Resource; RidsResOd Resource; RidsNmssOd Resource; RidsRgn1MailCenter Resource; RidsRgn2MailCenter Resource; RidsRgn3MailCenter Resource; RidsRgn4MailCenter Resource; RidsNrrDorlLpl3-2 Resource; RidsNrrPMByron Resource; RidsNrrPMBraidwood Resource; RidsNrrDss Resource; RidsNrrDe Resource; RidsNrrDpr Resource; RidsNrrDorl Resource; Garmoe, Alex; Keene, Todd; Gody, Tony; Gendelman, Adam; Mizuno, Beth; Correia, Richard; West, Khadijah; Bailey, Marissa; Scarbrough, Thomas; Spencer, Michael; Clark, Theresa
Subject:	OEDO-16-00463 Nuclear Energy Institute Comments in Support of Exelon Generation Company Second-Level Backfit Appeal

Date: September 15, 2016

Memorandum To: Anthony R. Pietrangelo

From: Victor M. McCree

Subject: OEDO-16-00463 Nuclear Energy Institute Comments in Support of Exelon Generation Company Second-Level Backfit Appeal

View ADAMS P8 Properties ML16208A006
Open ADAMS P8 Package (OEDO-16-00463 Nuclear Energy Institute Comments in Support of Exelon Generation Company Second-Level Backfit Appeal.)

U.S.NRC
United States Nuclear Regulatory Commission
Protecting People and the Environment

Ticket: OEDO-16-00463

General Information

Assigned Offices: OEDO
Other Parties:

OEDO Due Date: 10/30/2016
SECY Due Date:
OEDO Original Due Date: 10/30/2016
Date Response Requested by Originator: 10/30/2016

Subject: Nuclear Energy Institute Comments in Support of Exelon Generation Company Second-Level Backfit Appeal

Description:

CC Routing:

Incoming ADAMS Accession:
Incoming ADAMS Package:
Response Accession:

Other Information

Cross Reference No: Project Number 689
SRM: No

Process Information

Action Type: Letter
Signature Level: OEDO - EDO
Special Instructions:

OEDO Concurrence: No
OCM Concurrence: No
OCA Concurrence: No

Document Information

Originator Name: Anthony R. Pietrangelo
Originator Org.: Industry - NEI

Date of Incoming:
Document Received by OEDO Date:

Addressee Name: McCree V M
Addressee Affiliation: NRC/EDO
Incoming Task: Letter

OEDO POC: Clark, Theresa (txv)

From:	Lewis, LaShawnna
Sent:	Friday, September 16, 2016 12:05 PM
To:	Clark, Theresa; Wiebe, Joel
Cc:	Keene, Todd; Orf, Tracy; Rohrer, Shirley; Miller, Ed; Brown, Eva
Subject:	RE: REQUEST: listserv Exelon letter

Good Afternoon All,

Listserv Complete! Thanks.

Shawnna Lewis
Administrative Assistant
U.S. Nuclear Regulatory Commission
NRR/DORL/LPLIII-2 and LPLIV-2
OWFN 08-H4
301-415-1389
LaShawnna.Lewis@nrc.gov

From: Clark, Theresa
Sent: Friday, September 16, 2016 11:54 AM
To: Wiebe, Joel <Joel.Wiebe@nrc.gov>; Lewis, LaShawnna <LaShawnna.Lewis@nrc.gov>
Cc: Keene, Todd <Todd.Keene@nrc.gov>; Orf, Tracy <Tracy.Orf@nrc.gov>; Rohrer, Shirley <Shirley.Rohrer@nrc.gov>; Miller, Ed <Ed.Miller@nrc.gov>; Brown, Eva <Eva.Brown@nrc.gov>
Subject: RE: REQUEST: listserv Exelon letter

Thanks so much! Yes, it's been distributed internally already.

From: Wiebe, Joel
Sent: Friday, September 16, 2016 11:53 AM
To: Lewis, LaShawnna <LaShawnna.Lewis@nrc.gov>
Cc: Keene, Todd <Todd.Keene@nrc.gov>; Orf, Tracy <Tracy.Orf@nrc.gov>; Clark, Theresa <Theresa.Clark@nrc.gov>; Rohrer, Shirley <Shirley.Rohrer@nrc.gov>; Miller, Ed <Ed.Miller@nrc.gov>; Brown, Eva <Eva.Brown@nrc.gov>
Subject: RE: REQUEST: listserv Exelon letter

Lashawnna,

Can you listserve this? I think the rest of the dispatch will be done upstairs, but you may want to check with Theresa to verify that.

Just listserve it via the normal Byron/Braidwood listserve process.

Joel

From: Clark, Theresa
Sent: Friday, September 16, 2016 11:39 AM

From:	Clark, Theresa
Sent:	Sunday, September 11, 2016 8:24 PM
To:	McCree, Victor; Holahan, Gary
Cc:	Johnson, Michael; Tracy, Glenn
Subject:	RE: backfit documents

Done—thanks! I will print first thing in the morning; Gary and Mike can take a look tomorrow (if they haven't already), and after that all three documents will be ready to go once we have the NLO on the Exelon letter (expecting that tomorrow).

All 3 documents are publicly available in ADAMS.

Links again, for those on the network:

- **Letter responding to Exelon**
  View ADAMS P8 Properties ML16243A067
  Open ADAMS P8 Document (09/XX/16 Letter to Exelon from Victor McCree)
- **Letter responding to NEI**
  View ADAMS P8 Properties ML16246A150
  Open ADAMS P8 Document (09/XX/16 NEI Comments in Support of Exelon Generation Company Second Level Appeal (To: Anthony Pietrangelo, From: Victor McCree))
- **Memo to NRR**
  View ADAMS P8 Properties ML16246A247
  Open ADAMS P8 Document (Appeal of Backfit Imposed in Braidwood and Byron Stations (To: William Dean, From: Victor McCree))

--

Theresa Valentine Clark
Executive Technical Assistant (Reactors)
U.S. Nuclear Regulatory Commission
Theresa.Clark@nrc.gov | 301-415-4048 | O-16E22

From: McCree, Victor
Sent: Sunday, September 11, 2016 6:14 PM
To: Clark, Theresa <Theresa.Clark@nrc.gov>; Holahan, Gary <Gary.Holahan@nrc.gov>
Cc: Johnson, Michael <Michael.Johnson@nrc.gov>; Tracy, Glenn <Glenn.Tracy@nrc.gov>
Subject: RE: backfit documents

Please incorporate my final edits (attached). Thanks again!

Vic

From: Clark, Theresa
Sent: Saturday, September 10, 2016 9:32 PM
To: McCree, Victor <Victor.McCree@nrc.gov>; Holahan, Gary <Gary.Holahan@nrc.gov>
Subject: RE: backfit documents

Hi Vic!

I worked with Tom on an updated response to question 2.c that is much stronger/more specific. It along with the rest of your reclama response are now included in the enclosure to the memo to NRR, both in ADAMS and attached. The tracked version includes the new response alongside the old, as well as the light edits that I made (again, mostly acronym definitions).

Please let me know if you'd like any further changes to this or the other documents and I can have them ready by the time we need them Monday. Thanks!

View ADAMS P8 Properties ML16246A247
Open ADAMS P8 Document (Appeal of Backfit Imposed in Braidwood and Byron Stations (To: William Dean, From: Victor McCree))

--
Theresa Valentine Clark
Executive Technical Assistant (Reactors)
U.S. Nuclear Regulatory Commission
Theresa.Clark@nrc.gov | 301-415-4048 | O-16E22

From: McCree, Victor
Sent: Friday, September 09, 2016 4:46 PM
To: Clark, Theresa <Theresa.Clark@nrc.gov>; Holahan, Gary <Gary.Holahan@nrc.gov>
Subject: Re: backfit documents

Theresa/Gary,

See attached edits to the reclama response. I need a stronger response to question 2.c. Please work either Tom and let me know what you propose.

Thanks, Vic

From: Clark, Theresa
Sent: Friday, September 09, 2016 8:29 AM
To: Tracy, Glenn (Glenn.Tracy@nrc.gov)
Cc: Holahan, Gary
Subject: FYI: backfit appeal review panel 1-pager
Attachments: OEDO presentation 082416.docx

Glenn,

Per our discussion this morning, for your awareness, attached is the 1-pager that the Exelon backfit appeal review panel used to brief Vic and Mike on the panel's recommendation. Note that this of course does not represent Vic's decision on the matter, just the panel recommendation. The panel's full report is available in ADAMS at the following link.

View ADAMS P8 Properties ML16236A198
Open ADAMS P8 Package (Backfit Appeal Review Panel Findings (Byron and Braidwood))

--
Theresa Valentine Clark
Executive Technical Assistant (Reactors)
U.S. Nuclear Regulatory Commission
Theresa.Clark@nrc.gov | 301-415-4048 | O-16E22

OEDO Briefing on Exelon Backfit Appeal Review Panel Results
Gary Holahan | Steve West | Tom Scarbrough | Michael Spencer | Theresa Clark – August 24, 2016

What is the action?
EDO-level (second) appeal of backfit issued to Exelon for Byron and Braidwood (B/B) in October 2015.

What was the backfit about?
- The NRC staff determined B/B were not in compliance with the **plant-specific design bases, GDCs 15/21/29, and 10 CFR 50.34(b)**, related to **ANS-51.1/N18.2-1973** provisions on events progressing to more serious conditions (non-escalation).
- **Valve failure following water discharge** could cause escalation, predicated on several positions:
  - "water relief through a valve that is not qualified for water relief will cause that valve to stick in its fully open position" (emphasis added)
  - "the licensee ... has not applied the single-failure assumption" (emphasis added)
  - "nor have they provided ASME water qualification documentation ..." (emphasis added)
- Position differs from 5/14/01 SE supporting a stretch power uprate (referred to as the **Uprate SE**).
  - Staff determined backfitting justified under the **compliance exception** (10 CFR 50.109(a)(4)(i)) and the licensee was directed to take action to resolve the non-compliance.

What is (some of) the history?
- **1968-1972:** GDCs define AOOs (normal to once in plant life) and postulated accidents
- **1970+:** ANS (and Westinghouse) formulate ANS Conditions I (normal), II (frequent), III (infrequent), IV (accident) and non-escalation for transient analysis
- **1979-1980:** TMI Action Plan item II.D.1 requires "qualification" by testing of pressurizer valves
- **1993:** Westinghouse (NSAL-93-013) identifies analysis problems (no Part 21 or generic NRC action)
- **1996+:** Westinghouse PWRs update under 50.59 or request license amendments with reanalysis, PORV upgrades, safety valve crediting, etc. (B/B withdrew their request)
- **2001 and 2004:** NRR issues B/B amendments, including credit for safety valve water discharge
- **2005:** RIS-05-029 – PWR analyses include errors (e.g., non-safety PORVs, un-qualified valves)
- **2013:** Proposed RAI on a B/B measurement uncertainty uprate declared out of scope
- **2015+:** Backfit; appealed to NRR and upheld in 2016, calling for ASME qualification of safety valves, application of single failure criterion, assumption of failure on water discharge; re-appealed to EDO

What did the appeal review panel do?
- Reviewed over 100 documents
- Interviewed NRR (systems and engineering), OGC, 2001 reactor systems supervisor, CRGR
- Reviewed NRR response to preliminary findings
- Obtained risk insights from RES

What did the appeal review panel conclude?
- EPRI testing (TMI Action Plan Item II.D.1) showed that valves did not fail open on water discharge
- Single failure criterion has not been traditionally applied to safety valves sticking open
- ASME certification has not been required to credit safety valves' water discharge capability
- Event escalation and compliance with 10 CFR 50.34(b) and GDCs are therefore no longer at issue.

What does the appeal review panel recommend?
- EDO favorable response to Exelon compliance backfit appeal
- Direct NRR to verify that all PWRs resolved issue, re-evaluate RIS-05-029 and the draft RIS-05-029 revision 1
- Note that generic insights are being sent to CRGR for consideration in addressing tasking memo

Answers to EDO's Questions in Charter:

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

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

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

   Answer: The standard in place in 2001 and 2004 and at present is that failures of pressurizer safety valves to reclose need not be assumed to occur following water discharge if the likelihood is sufficiently small, based on well-informed staff engineering judgment.

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

   Answer: For Byron and Braidwood, the standard for progression of postulated events between categories of severity is set forth in the Updated Final Safety Analysis Report (UFSAR), as described in the NRC staff's October 9, 2015, backfit imposition letter. The Panel supports the NRC staff's view that non-escalation (from Condition II to Condition III or IV, as defined in ANS-51.1/N18.2-1973) is a known and established standard applicable to Byron and Braidwood. However, this event progression standard does not establish specific standards for valve qualification. Therefore, it is not the basis for a compliance backfit given this set of facts. Regulatory Issue Summary 2005-29 and its draft Revision 1 do not alter this conclusion.

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

   Answer: For the specific technical issues reviewed by the Panel, the Panel concluded that the current licensing basis for B/B complies with the applicable regulations and provides adequate protection of the public health and safety.

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

   Answer: The RES analysis provides insights on the risk significance of the sequence at issue. This analysis suggests that an inadvertent ECCS actuation sequence, assuming that pressurizer overfill would lead to a small loss-of-coolant accident, contributes approximately 1 percent of the total internal events CDF. If the backfit were implemented such that the pressurizer safety valves would always reclose properly, the CDF reduction is estimated at 1.5E-07 per year. If the pressurizer safety valves 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 appeal Panel, is responsible for any decisions on alternative application of the backfit rule to this issue. Second, the Panel does not wish to imply that "the contribution to plant risk" should be seen as the only measure of enhanced safety. For example, defense-in-depth has a recognized role and value in the regulatory process.

From: Holahan, Gary
Sent: Thursday, July 14, 2016 1:20 PM
To: Correia, Richard
Cc: West, Steven; Clark, Theresa; Scarbrough, Thomas; Spencer, Michael
Subject: Millstone 3 inadvertent ECCS injection
Attachments: 2016 Backfit Panel - Millstone event Special Inspection July 5 2005.pdf

... report attached

This attachment is publicly available in ADAMS as ML051860338.

From: Holahan, Gary
Sent: Wednesday, August 03, 2016 5:18 PM
To: Clark, Theresa
Subject: RE: Things to review

(b)(6) See you tomorrow, hopefully.

...and let me be the judge of how important your contributions are.

From: Clark, Theresa
Sent: Wednesday, August 03, 2016 5:10 PM
To: Holahan, Gary <Gary.Holahan@nrc.gov>
Subject: Re: Things to review

Thanks. I hope to be back tomorrow (left today when (b)(6) and will read it then.

I appreciate the promotion and all, but I certainly didn't read the documents in the detail that you guys did... So I would be happy to be on concurrence (however Margie K was for the Hatch one) but don't need to be on the front of the memo unless you think it's super important.

Theresa

On: 03 August 2016 16:58, "Holahan, Gary" <Gary.Holahan@nrc.gov> wrote:

Steve,
Tom,
Michael,
Theresa,

I have taken the Preliminary Findings document and incorporated it into a "Discussion" section. I added text to make it read like a report (no changes to findings or conclusions). Please review at Reports / Backfit Appeal Report 2016 08 03 2pm.

Next I will start on the Enclosures (and the sections referring to them).

I have also drafted (first draft...) a memo to Vic presenting the report. Please review at Reports / Cover memo Backfit Appeal Panel 2016 08 03 2pm.

Gary

P.S. I told Vic that I promoted you to team member, so he won't be surprise to see your name on the cover memo.

From: West, Steven
Sent: Tuesday, August 23, 2016 12:28 PM
To: Clark, Theresa
Subject: RE: ACTION: package for panel report

I ran the report you attached to this email through the "inspect document" feature and it found some embedded stuff.

Steve

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

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

From: Clark, Theresa
Sent: Tuesday, August 23, 2016 12:17 PM
To: West, Steven <Steven.West@nrc.gov>
Subject: Re: ACTION: package for panel report

Hope so--but I did accept all edits and delete all comments, so there's that.

On: 23 August 2016 12:10, "West, Steven" <Steven.West@nrc.gov> wrote:
I assume she's also going to run the process where Word checks the documents for hidden comments, text, etc. and strips them out of the final documents.

Steve

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

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

From: Clark, Theresa
Sent: Tuesday, August 23, 2016 12:08 PM
To: Spencer, Michael <Michael.Spencer@nrc.gov>

Cc: Holahan, Gary <Gary.Holahan@nrc.gov>; West, Steven <Steven.West@nrc.gov>; Scarbrough, Thomas <Thomas.Scarbrough@nrc.gov>
Subject: Re: ACTION: package for panel report

I knew I was supposed to do something else in the cover memo! You get the prize :).

I'll put those in as soon as Patti is done.

Thanks!!

On: 23 August 2016 12:05, "Spencer, Michael" <Michael.Spencer@nrc.gov> wrote:
Theresa, I looked at the updated documents.

On the report: Page 5 has a "that that."

The cover memo (page 2) has the following reference to the NRR panel member that I thought we were going to get rid of: "The Panel notes, as did a member of the earlier NRR backfit appeal panel (ADAMS Accession No. ML16081A405), that the issue of pressurizer valve performance following water discharge appears to have generic applicability, and is not specific to Byron and Braidwood."

I have no other comments, but I will continue the typo check of the references (haven't found any yet).

Michael

From: Clark, Theresa
Sent: Tuesday, August 23, 2016 11:50 AM
To: Sprogeris, Patricia <Patricia.Sprogeris@nrc.gov>
Cc: Holahan, Gary <Gary.Holahan@nrc.gov>; West, Steven <Steven.West@nrc.gov>; Scarbrough, Thomas <Thomas.Scarbrough@nrc.gov>; Spencer, Michael <Michael.Spencer@nrc.gov>
Subject: ACTION: package for panel report
Importance: High

Patti,

Could you please create a package in ADAMS and add the two attached documents, then prepare a paper concurrence package? (It does not need to be pinked, just printed and assembled.) Feel free to fix formatting (e.g., margin for letterhead) but please do not trouble yourself with editing as we have already gone over it many times. Please do add the ML#s on the front cover of the report file and on the concurrence page of the memo file.

I need this sometime this afternoon if possible.

Panel folks—FYI. Prizes for anyone who finds an error at this point ☺. I will let you know once I have a package ready for signature (and we can work in any concurrence edits there).

THANKS!!

--
Theresa Valentine Clark
Executive Technical Assistant (Reactors)
U.S. Nuclear Regulatory Commission
Theresa.Clark@nrc.gov | 301-415-4048 | O-16E22

From:	West, Steven
Sent:	Wednesday, August 24, 2016 12:52 PM
To:	Hackett, Edwin
Cc:	Holahan, Gary; Clark, Theresa
Subject:	FW: Exelon Backfit Appeal Panel Report
Attachments:	2016-08-23 Final Backfit Appeal Memo (MASTER) ML16236A202.docx; 2016-08-23 Final Backfit Appeal Panel Report (MASTER) ML16236A208.docx

These two attachments are publicly available in ADAMS.

Ed,

For your awareness. I've attached the report for your convenience. This is not public, but you can distribute to the CRGR members for information if you'd like.

Steve

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

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

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

NRR,

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

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

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

Gary

From: McCree, Victor
Sent: Sunday, September 11, 2016 12:13 PM
To: Clark, Theresa; Holahan, Gary
Subject: Re: backfit documents

Outstanding, thanks Theresa! We're almost there....

Vic

On: 10 September 2016 21:32, "Clark, Theresa" <Theresa.Clark@nrc.gov> wrote:
Hi Vic!

I worked with Tom on an updated response to question 2.c that is much stronger/more specific. It along with the rest of your reclama response are now included in the enclosure to the memo to NRR, both in ADAMS and attached. The tracked version includes the new response alongside the old, as well as the light edits that I made (again, mostly acronym definitions).

Please let me know if you'd like any further changes to this or the other documents and I can have them ready by the time we need them Monday. Thanks!

View ADAMS P8 Properties ML16246A247
Open ADAMS P8 Document (Appeal of Backfit Imposed in Braidwood and Byron Stations (To: William Dean, From: Victor McCree))

--
Theresa Valentine Clark
Executive Technical Assistant (Reactors)
U.S. Nuclear Regulatory Commission
Theresa.Clark@nrc.gov | 301-415-4048 | O-16E22

From: McCree, Victor
Sent: Friday, September 09, 2016 4:46 PM
To: Clark, Theresa <Theresa.Clark@nrc.gov>; Holahan, Gary <Gary.Holahan@nrc.gov>
Subject: Re: backfit documents

Theresa/Gary,

See attached edits to the reclama response. I need a stronger response to question 2.c. Please work either Tom and let me know what you propose.

Thanks, Vic

From:	Clark, Theresa
Sent:	Wednesday, August 03, 2016 6:39 PM
To:	Whitman, Jennifer
Subject:	Re: Exelon Backfit Appeal Panel Preliminary Findings FOR COMMENT - OUO- Pre-decisional - Internal NRC Use Only -

Yes--I was in the process of putting together a nicely formatted list. I'll send you what I have in the morning (assuming I'm in, (b)(6) today--if I'm out I'll ask Gary to send) and can give you any specific ones that you didn't see on there.

Thanks,
Theresa

On: 03 August 2016 18:17, "Whitman, Jennifer" <Jennifer.Whitman@nrc.gov> wrote:
Theresa,

Do you have ML numbers or electronic copies of the references that are in the findings report?

Thanks,

Jen

From: Mcginty, Tim
Sent: Monday, August 01, 2016 6:14 PM
To: Whitman, Jennifer <Jennifer.Whitman@nrc.gov>; Oesterle, Eric <Eric.Oesterle@nrc.gov>; Hickey, James <James.Hickey@nrc.gov>; Taylor, Robert <Robert.Taylor@nrc.gov>
Subject: FW: Exelon Backfit Appeal Panel Preliminary Findings FOR COMMENT - OUO- Pre-decisional - Internal NRC Use Only -

Team – please evaluate per the panel's request for any comments or clarifications, etc. Note that any comments provided will be reflected in the final recommendations, and ultimately I would anticipate being made publically available. I would think that we will want to meet on this in the near future.

Please also take your usual care to treat the information as OUO pre-decisional and internal use only. Thanks,
Tim

From: Holahan, Gary
Sent: Monday, August 01, 2016 5:57 PM
To: Dean, Bill <Bill.Dean@nrc.gov>; Lubinski, John <John.Lubinski@nrc.gov>; Mcginty, Tim <Tim.McGinty@nrc.gov>; Akstulewicz, Frank <Frank.Akstulewicz@nrc.gov>; Doane, Margaret <Margaret.Doane@nrc.gov>; Mcdermott, Brian <Brian.McDermott@nrc.gov>; Bailey, Marissa <Marissa.Bailey@nrc.gov>
Cc: Hackett, Edwin <Edwin.Hackett@nrc.gov>; West, Steven <Steven.West@nrc.gov>; Clark, Theresa <Theresa.Clark@nrc.gov>; Scarbrough, Thomas <Thomas.Scarbrough@nrc.gov>; Spencer, Michael <Michael.Spencer@nrc.gov>; Evans, Michele <Michele.Evans@nrc.gov>; Williamson, Edward <Edward.Williamson@nrc.gov>; Mizuno, Geary <Geary.Mizuno@nrc.gov>; Shuaibi, Mohammed <Mohammed.Shuaibi@nrc.gov>; Mccree, Victor <Victor.McCree@nrc.gov>; Johnson, Michael

<Michael.Johnson@nrc.gov>; Tracy, Glenn <Glenn.Tracy@nrc.gov>; Gody, Tony <Tony.Gody@nrc.gov>
Subject: Exelon Backfit Appeal Panel Preliminary Findings FOR COMMENT - OUO- Pre-decisional - Internal NRC Use Only
-

All,

Consistent with the plan we presented last week, attached are the preliminary findings of the Exelon Backfit Appeal Panel. The Summary from the Preliminary Findings is reproduced below. The preliminary findings were discussed briefly with the OEDO for their awareness.

As indicated in our completion plan, the panel would appreciate any comments on, or additions to: the documents sited; their interpretation and intent; or the understanding of the backfit rule compliance exception. Comments would be appreciated by August 9, 2016, but can be accepted as last as August 15, 2016. The panel will also be available for discussion any time before August 15, 2016.

Comments will be reflected or acknowledged in the panel's final report and recommendations to the EDO.

The Preliminary Findings document attached is an internal, pre-decisional document at this time. Both Exelon and NEI declined offers for a public meeting on this issue.

Gary ... for the panel
-Steve West
-Tom Scarborough
-Michael Spencer
-Theresa Clark

In summary:

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

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

However, none of these positions were "known and established standards of the Commission" in 2001 or 2004 for determining when it was appropriate to assume a failure of PSVs to reseat. In fact, they were not "known and established standards of the Commission" in 2005 or 2006 or 2007.

Moreover, two of these positions do not appear to be "established standards of the Commission" at present, since the call for use of the single failure criterion first appears in proposed 2015 draft Revision 1 to RIS 2005-029, and the call for ASME certification first appears in the Exelon compliance backfit. The panel concludes that the standard in place in 2001 and 2004 and at present is simply that the probability of failure of a Pressurizer Safety Valve (PSV) is sufficiently small, based on well-informed staff engineering judgement, and that the use of the word "qualified" or "qualification" implied only a general demonstration of capability, such as in the EPRI testing done in response to TMI Action Plan Item II.D.1.

The panel concludes that, in 2001 and 2004, the staff was not misinformed nor did it "err" in approving the Byron and Braidwood power uprates ... nor was it in error in approving other similar cases (e.g. Beaver Valley in 2006). The 2015 staff positions taken to support the compliance backfit finding represent new and different staff views on how to address potential PSV failures following water discharge. Although they represent well-intentioned staff positions that could provide additional safety margin, they do not provide a basis for a compliance backfit.

The panel's findings therefore support the Exelon backfit appeal.

In addition to the specific finding relating to the backfit appeal, the panel believes it is important to acknowledge 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. The panel concludes this while fully aware that the event sequence being considered appears to be of little safety significance (the panel has requested RES analysis to confirm this belief). Operator training and emergency procedures to terminate the event before pressurizer filling, as well as the use of power-operated relief valves rather than relying solely on PSVs, are clearly preferred, whether they form the facilities' UFSAR licensing basis or not.

The panel has not (at this time) formed any views on whether a backfit on this topic could be justified as "adequate protection" or "cost justified"; or whether a "forward-fit" staff position is appropriate or not.

From: West, Steven
Sent: Tuesday, July 05, 2016 4:12 PM
To: Holahan, Gary; Scarbrough, Thomas; Spencer, Michael
Cc: Clark, Theresa
Subject: Staff responses to public comments on draft RIS 2005-29, Rev. 1
Attachments: Comment Resolution Memo NLO.docx

The attachment is publicly available in ADAMS as ML15264A283.

Attached fyi.

Steve

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

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

From:	Whitman, Jennifer
Sent:	Tuesday, July 19, 2016 3:03 PM
To:	Clark, Theresa
Cc:	Mcginty, Tim; Holahan, Gary; West, Steven; Scarbrough, Thomas; Spencer, Michael
Subject:	RE: Exelon backfit discussion
Attachments:	SRXB RAIS for Byron and Braidwood MUR accident analysis

I do not have either of those two EPRI reports. I attached the RAIs that SRXB wanted to send on the MUR and this is the link to the non-concurrence that was filed on the SE for the MUR by Sam.

Jen

From: Clark, Theresa
Sent: Monday, July 18, 2016 1:57 PM
To: Whitman, Jennifer <Jennifer.Whitman@nrc.gov>
Cc: Mcginty, Tim <Tim.McGinty@nrc.gov>; Holahan, Gary <Gary.Holahan@nrc.gov>; West, Steven <Steven.West@nrc.gov>; Scarbrough, Thomas <Thomas.Scarbrough@nrc.gov>; Spencer, Michael <Michael.Spencer@nrc.gov>
Subject: RE: Exelon backfit discussion

Hi there! As promised, here are the microfiche references from the earlier Byron/Braidwood amendment requests we were discussing today. When Gary mentioned the printouts I forgot I had already saved copies from the scanner in the library. So, you get the copies too ☺ -- and as long as we don't add this version to ADAMS vs the official copy the records people don't get annoyed. The filenames show the microfiche addresses in case you need them.

- 12/9/1997 letter providing an amendment supplement identifying an issue with the spurious SI analysis
- 12/19/1997 letter providing path forward on spurious SI (planning amendment on automatic action of PORVs)
- 5/29/1998 LAR regarding automatic action of PORVs
- 5/13/1999 electrical RAI on the LAR above
- 7/16/1999 withdrawal of LAR above and RAI response

If you come across EPRI NP-2770 Volume 6 (Crosby safety valves) or NP-2670 (PORV test reports), please let us know. Thanks again!

--
Theresa Valentine Clark
Executive Technical Assistant (Reactors)
U.S. Nuclear Regulatory Commission
Theresa.Clark@nrc.gov | 301-415-4048 | O-16E22

-----Original Appointment-----
From: Clark, Theresa
Sent: Thursday, July 14, 2016 5:08 PM
To: Clark, Theresa; Mcginty, Tim; DSSCAL Resource; Holahan, Gary; West, Steven; Scarbrough, Thomas; Spencer, Michael
Cc: Whitman, Jennifer
Subject: Exelon backfit discussion
When: Monday, July 18, 2016 12:30 PM-1:30 PM (UTC-05:00) Eastern Time (US & Canada).
Where: HQ-OWFN-16B06-12p

Hi Tim,

Thanks for being willing to meet with the EDO's appeal panel for the Exelon backfit. As we discussed on the phone, you can bring staff if you would like to. However, you may not feel the need at this point—we are intending for this to be a casual conversation about the technical issues that led to the backfit and aren't sending any preparatory materials/questions. If we need further discussions (e.g., with particular staff) after this we can certainly do that.

Also—I know this isn't a great time (and Steve has a potential conflict) but getting another time in the next two weeks was nigh on impossible. Let me know if it is really bad timing for you. Thanks!

Background References:

- Appeal panel charter: ML16173A311
- 6/16/16 NEI letter supporting Exelon backfit appeal to EDO: [attached, not yet in ADAMS]
- 6/2/16 Exelon backfit appeal to EDO: ML16154A254
- 5/3/16 NRR backfit appeal decision: ML16095A204
- 12/8/2015 Exelon backfit appeal to NRR: ML15342A112
- 10/9/2015 NRC backfit letter: ML14225A871
- 8/26/04 pressurizer safety valve setpoint safety evaluation: ML042250531
- 5/4/01 stretch power uprate safety evaluation: ML033040016

The NEI letter is publicly available in ADAMS as ML16246A150.

All other documents are publicly available in ADAMS under the specified accession numbers.

From: Correia, Richard
Sent: Wednesday, July 20, 2016 10:03 AM
To: Holahan, Gary; West, Steven; Clark, Theresa
Cc: Weber, Michael; Hackett, Edwin; Thaggard, Mark
Subject: Plan for B/B Risk Perspective Task
Attachments: Talking Points - DRA internal - 072016.docx

Gary, Steve, Theresa,

Following our meeting last week on the Byron/Braidwood appeal, my staff developed a risk analysis plan to evaluate a relief valve failure causing a Condition II event to become a Condition III/IV event. The plan is attached for your information and comment.

Best

Rich

Richard P. Correia, P.E.
Director, Division of Risk Analysis
Office of Nuclear Regulatory Research
U.S. NRC

July 19, 2016
RES/DRA

OVERVIEW

Tasking: Develop quick-turn-around risk perspective on relief valve failure causing a Condition II event to become a Condition III/IV event.

Backdrop: Byron & Braidwood backfit appeal related to safety analysis assumptions for Condition II events (e.g., inadvertent safety injection)

Framing the question:

- What can go wrong?: A Condition II anticipated operational occurrence can lead to a stuck-open relief valve, in turn leading to a Condition III/IV loss-of-coolant accident
- How likely is it to occur?: TBD – this is the conditional probability of the transient degrading in to a LOCA
- What are the consequences?: TBD – this is the increase in CDF

General assumptions:

- Circa 2010 procedures (readily available to RES) continue to apply
- Utilize existing SPAR model with minor upgrades
- Assume all 4 units are identical for the purposes of this analysis
- Risk exists from events that lead to steam or water lifts of pressurizer PORVs or pressurizer SRVs (i.e., problem is framed more broadly than just SRV water lifts) due to potential for relief valves to fail open following passing steam or water
- In general, once an SRV or PORV fails open, a Condition III/IV event exists:
    - No consideration of leakage size or downstream piping failures beyond what is inherent in the valve failure probabilities;
    - We will consider potential operator actions to recover a failed-open PORV by closing the associated PORV block valve(s).
- In addition to human failure events included in the base SPAR model, termination of inadvertent injection from CVCS or ECCS will also be evaluated (including consideration of time availability)

Near-term needs from entities external to RES:

- Feedback on this approach (Appeal Review Board)
- Information on Byron/Braidwood's typical PORV block valve usage (R-III)

July 19, 2016
RES/DRA

WORK PLAN

Phase 1: 2 Weeks.

1. Align with INL on which model is best to use. Task INL with adding inadvertent SI, very small LOCA, and steam line break event trees to the SPAR model. Use existing frequencies/probabilities. Use 0.1 screening value for new operator action failure probabilities (unless a basis for an alternative value is readily available).

2. Obtain plant configuration information (e.g., how the plant is normally operated with respect to how many block valves are closed). Obtain/develop information to assess operator action timing, including EOP pathway review.

3. Assemble a list and values of key critical parameters, success criteria, probabilities, assumptions, and operator actions (e.g., simple time estimates for pressurizer overfill). As a first estimate, use readily-available information and values. Review operating experience on valve performance.

4. Quantify baseline CDF cases, and perform sensitivities as time permits and available information warrants (example table provided below).

	Inadvertent SI sequence(s) involving SRV passing of water (% of CDF)			All transient sequences involving stuck-open relief valves (% of CDF)		
	Base model	Human reliability sensitivity	Valve reliability sensitivity	Base model	Human reliability sensitivity	Valve reliability sensitivity
0 PORVs blocked						
1 PORV blocked						
2 PORVs blocked						

5. Document results and discuss their implications.

6. Have HRA expert reflect on HFE characterization and HEP quantification.

❖ *Deliverable: Brief summary report describing approach and results.*

Phase 2: 2 Weeks.

7. Based on results of Phase 1, further work could be done, as needed.

From: West, Steven
Sent: Tuesday, August 16, 2016 3:01 PM
To: Clark, Theresa
Subject: RE: Final Report

Theresa,

Thx. We're still meeting tomorrow at 8:30? Call me a (b)(6)

Steve

-------- Original Message --------
From: "Clark, Theresa" <Theresa.Clark@nrc.gov>
Date: Tue, August 16, 2016 2:43 PM -0400
To: "West, Steven" <Steven.West@nrc.gov>, "Holahan, Gary" <Gary.Holahan@nrc.gov>, "Scarbrough, Thomas" <Thomas.Scarbrough@nrc.gov>, "Spencer, Michael" <Michael.Spencer@nrc.gov>
Subject: RE: Final Report

Here you go. Please note that I will be editing and updating the references over the next couple of days.

From: West, Steven
Sent: Tuesday, August 16, 2016 2:20 PM
To: Holahan, Gary <Gary.Holahan@nrc.gov>; Clark, Theresa <Theresa.Clark@nrc.gov>; Scarbrough, Thomas <Thomas.Scarbrough@nrc.gov>; Spencer, Michael <Michael.Spencer@nrc.gov>
Subject: Re: Final Report

Theresa,

Please email to me the docs that Gary wants us to look at. I have time to review them while on travel, but have last Friday's versions with me. Thanks much.

Steve

-------- Original Message --------
From: "Holahan, Gary" <Gary.Holahan@nrc.gov>
Date: Tue, August 16, 2016 1:31 PM -0400
To: "West, Steven" <Steven.West@nrc.gov>, "Clark, Theresa" <Theresa.Clark@nrc.gov>, "Scarbrough, Thomas" <Thomas.Scarbrough@nrc.gov>, "Spencer, Michael" <Michael.Spencer@nrc.gov>
Subject: Final Report

Steve,
Tom,
Michael,
Theresa,

After reviewing and re-reviewing the NRR comments, I made several additions to the cover letter and the main report. I believe it is ready for (almost) final polishing.

Please review the cover "Master" and the report "Master" in the report section. I would like to aim for Thursday or Friday (I'm out Friday) for completion. That will give us a few days margin to the due date.

Thanks,

Gary

From: Scarbrough, Thomas
Sent: Tuesday, August 16, 2016 5:04 PM
To: Holahan, Gary
Cc: West, Steven; Clark, Theresa; Spencer, Michael
Subject: RE: Another version of section 5

Gary,

How about something like the following?

ADDITIONAL PANEL THOUGHTS

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

Operator training, and Emergency Operating Procedures to terminate the event before pressurizer filling, as well as the 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 Safety Valves in question were designed for steam service. Steam relief is their normal service condition; and applies to their certification. The Panel supports the previous staff determinations for Braidwood/Byron and certain other plants that PSVs experiencing water relief during an abnormal or accident condition need not be assumed to fail if there is a reasonable and technically well-informed engineering judgement to the contrary. However, the Panel also considers the actions by various licensees and accepted by the staff to improve the reliability and performance of the PORVs to avoid water relief through the PSVs to be prudent in light of their design specifications.

The Panel considered but could not determine the extent to which the Braidwood/Byron licensee addressed crediting water relief through the PSVs, PORVs, or PORV block valves in the Braidwood and Byron Inservice Test Programs.

The Panel notes that water relief through various pressurizer valves is not a new issue because water relief has always been credited (by the Braidwood/Byron licensee and other licensees) for the UFSAR Feedwater System Pipe Break analysis (15.2.8).

The Panel believes these thoughts to be worthwhile for consideration by the NRC staff and Braidwood/Byron licensee.

Thanks.

Tom

From: Holahan, Gary
Sent: Tuesday, August 16, 2016 3:37 PM
To: Scarbrough, Thomas <Thomas.Scarbrough@nrc.gov>
Cc: West, Steven <Steven.West@nrc.gov>; Clark, Theresa <Theresa.Clark@nrc.gov>; Spencer, Michael <Michael.Spencer@nrc.gov>
Subject: Another version of section 5

Tom,

I looking for a way to address the ITS question without the Appendix Conclusions and the "should" paragraphs.

See what you think of an alternate section 5, below?

Gary

ADDITIONAL PANEL THOUGHTS

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

Operator training, and Emergency Operating Procedures to terminate the event before pressurizer filling, as well as the 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 Safety Valves in question were designed for steam discharge. Steam discharge is their normal service condition; and it is what they were certified for. The panel supports the earlier staff judgments that: PSVs when experiencing conditions beyond their normal design conditions need not be assumed to fail if there is a reasonable and technically well-informed engineering judgement to the contrary.

The panel considered but could not determine how or if the licensee addressed crediting water discharge through the PSV in the Byron and Braidwood Inservice Test Program in conformance with ASME OM Code Case OMN-1. The panel notes that this is not a new issue since water discharge has always been credited (by the licensee and other licensees) for the UFSAR Feedwater System Pipe Break analysis (15.2.8).

From: West, Steven
Sent: Monday, August 22, 2016 3:29 PM
To: Scarbrough, Thomas; Clark, Theresa
Cc: Spencer, Michael; Holahan, Gary
Subject: RE: My comments on Friday's clean master

I like and support Tom's changes.

Steve

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

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

-----Original Message-----
From: Scarbrough, Thomas
Sent: Monday, August 22, 2016 12:19 PM
To: Clark, Theresa <Theresa.Clark@nrc.gov>
Cc: Spencer, Michael <Michael.Spencer@nrc.gov>; West, Steven <Steven.West@nrc.gov>; Holahan, Gary <Gary.Holahan@nrc.gov>
Subject: RE: My comments on Friday's clean master

Theresa,

In my attached markup of the Friday version of the report, I propose a few minor changes with comments in the margin.

My only significant suggestion regarding the report findings is my proposed ending to the sentence at the top of page 13 regarding the assumption that PSV failure following water discharge is a passive failure. I think we should add a provision to the end of the sentence that the licensee needs to justify that assumption (such as by EPRI testing).

The changes in the version provided with your e-mail this morning look fine with a few edits as follows:

- Footnote 13 has an extra "not" in the sentence.
- On page 4 in the second line of the second paragraph, the word "depended" appears misspelled.
- On page 8 in the last paragraph, the word "panel" should be capitalized.
- On page 18 in the first full paragraph, the second sentence should use "had" rather than "would" before "conducted"
- On page 21 in the last full paragraph, it appears that the last sentence should specify "prevent" rather than "ensure" pressurizer overfilling.
- On page 23 in the first paragraph, the last sentence has an extra "not"

Thanks.
Tom

-----Original Message-----
From: Clark, Theresa
Sent: Monday, August 22, 2016 8:44 AM
To: West, Steven <Steven.West@nrc.gov>; Holahan, Gary <Gary.Holahan@nrc.gov>
Cc: Spencer, Michael <Michael.Spencer@nrc.gov>; Scarbrough, Thomas <Thomas.Scarbrough@nrc.gov>
Subject: RE: My comments on Friday's clean master

Thank you, Steve!

I put in your comments through Appendix A and will be out of the office at an OEDO meeting most of the rest of the day. I'll get the rest in as soon as I can and resend.

Note that I put a few comments in the margin, mostly for Steve's awareness, but one for Michael to check and one as a placeholder based on an email from Steve (will address more later).

-----Original Message-----
From: West, Steven
Sent: Sunday, August 21, 2016 7:35 PM
To: Clark, Theresa <Theresa.Clark@nrc.gov>; Holahan, Gary <Gary.Holahan@nrc.gov>
Cc: Spencer, Michael <Michael.Spencer@nrc.gov>; Scarbrough, Thomas <Thomas.Scarbrough@nrc.gov>
Subject: My comments on Friday's clean master

Any thoughts on meeting again before Wed?

From: West, Steven
Sent: Monday, August 22, 2016 4:12 PM
To: Clark, Theresa
Cc: Spencer, Michael; Scarbrough, Thomas; Holahan, Gary
Subject: RE: My comments on Friday's clean master

Thanks, Theresa, for all of your timely and outstanding work on the report. I agree with all of the changes you made to the body of the report, as well as your explanatory comments. With respect to interactions vs meetings with staff, I had not considered in my comments the distinction you described. Whatever you decide is fine with me. Perhaps for meetings, we say meetings and for everything else, we say interactions. Also, your rule for handling tense works for me.

Looking forward to seeing the final appendices one more time.

One final thought/question: Since the report may become separated from the cover memo, should we identify the Panel members in the report itself?

Steve

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

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

-----Original Message-----
From: Clark, Theresa
Sent: Monday, August 22, 2016 8:44 AM
To: West, Steven <Steven.West@nrc.gov>; Holahan, Gary <Gary.Holahan@nrc.gov>
Cc: Spencer, Michael <Michael.Spencer@nrc.gov>; Scarbrough, Thomas <Thomas.Scarbrough@nrc.gov>
Subject: RE: My comments on Friday's clean master

Thank you, Steve!

I put in your comments through Appendix A and will be out of the office at an OEDO meeting most of the rest of the day. I'll get the rest in as soon as I can and resend.

Note that I put a few comments in the margin, mostly for Steve's awareness, but one for Michael to check and one as a placeholder based on an email from Steve (will address more later).

-----Original Message-----
From: West, Steven
Sent: Sunday, August 21, 2016 7:35 PM
To: Clark, Theresa <Theresa.Clark@nrc.gov>; Holahan, Gary <Gary.Holahan@nrc.gov>

Cc: Spencer, Michael <Michael.Spencer@nrc.gov>; Scarbrough, Thomas <Thomas.Scarbrough@nrc.gov>
Subject: My comments on Friday's clean master

Any thoughts on meeting again before Wed?

From: Scarbrough, Thomas
Sent: Wednesday, August 24, 2016 5:47 PM
To: Clark, Theresa
Cc: Holahan, Gary; West, Steven; Spencer, Michael
Subject: EPRI Valve Testing Locations

Theresa,

As a follow-up action from our meeting with the EDO today, I checked the various locations for the EPRI 1981 valve testing program.

According to EPRI NP-2628 (December 1982), the valve tests were performed in the following locations:

Safety Valve tests: EPRI/C-E Test Facility (Windsor, CT)

PORV tests: Wyle (Norco, CA) and Marshall Steam Station (Terrell, NC)

Thanks.
Tom

From: Coyne, Kevin
Sent: Friday, September 16, 2016 8:32 AM
To: Correia, Richard; Clark, Theresa
Cc: Sprogeris, Patricia
Subject: RE: Memorandum From: V. McCree to G. Holahan re: Charter for Backfit Appeal Review Panel Associated with Byron and Braidwood Compliance with 10 CFR 50.34(B), GDC 15, GDC 21, GDC 29, and the Licensing Basis

Importance: High

Hi Theresa –

We just had the header on there until the EDO's Office was ready to release the document. The version in ADAMS no longer has the "OUO internal information" header and the ADAMS profile title has been expanded. The document is still profiled as non-public. Do you want us to declare it publicly or is that something you plan to do with the other documents for the appeal review?

Kevin

Kevin Coyne, Ph.D., P.E.
Chief, Probabilistic Risk Assessment Branch
Division of Risk Analysis, Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001
(301) 415-2478 (work)

From: Correia, Richard
Sent: Friday, September 16, 2016 6:36 AM
To: Clark, Theresa <Theresa.Clark@nrc.gov>
Cc: Coyne, Kevin <Kevin.Coyne@nrc.gov>; Sprogeris, Patricia <Patricia.Sprogeris@nrc.gov>
Subject: RE: Memorandum From: V. McCree to G. Holahan re: Charter for Backfit Appeal Review Panel Associated with Byron and Braidwood Compliance with 10 CFR 50.34(B), GDC 15, GDC 21, GDC 29, and the Licensing Basis

We'll get this fixed Theresa asap.

Rich

Richard P. Correia, P.E.
Director, Division of Risk Analysis
Office of Nuclear Regulatory Research
U.S. NRC

From: Clark, Theresa
Sent: Thursday, September 15, 2016 5:26 PM

To: Correia, Richard <Richard.Correia@nrc.gov>
Cc: Coyne, Kevin <Kevin.Coyne@nrc.gov>; Sprogeris, Patricia <Patricia.Sprogeris@nrc.gov>
Subject: RE: Memorandum From: V. McCree to G. Holahan re: Charter for Backfit Appeal Review Panel Associated with Byron and Braidwood Compliance with 10 CFR 50.34(B), GDC 15, GDC 21, GDC 29, and the Licensing Basis

Rich,

We were getting ready to process this file for public release when I opened it to update the title in ADAMS, and now I see that it has OUO-Sensitive headers on it. I immediately unfiled it from the package that was going into DPC processing (it was not yet declared) and changed the availability back to non-publicly available.

Could you guys please look at the document (ML16214A199) and follow whatever processes you normally do to ensure that the content is OK, remove headers, complete the ADAMS profile (e.g., full title) and have it publicly released? At this point I am concerned that if I do it myself I will mess something up.

View ADAMS P8 Properties ML16214A199
Open ADAMS P8 Document (An Assessment of CDF.docx)

Thanks for your help!

--
Theresa Valentine Clark
Executive Technical Assistant (Reactors)
U.S. Nuclear Regulatory Commission
Theresa.Clark@nrc.gov | 301-415-4048 | O-16E22

From: Correia, Richard
Sent: Thursday, September 15, 2016 7:09 AM
To: Clark, Theresa <Theresa.Clark@nrc.gov>
Cc: Coyne, Kevin <Kevin.Coyne@nrc.gov>
Subject: RE: Memorandum From: V. McCree to G. Holahan re: Charter for Backfit Appeal Review Panel Associated with Byron and Braidwood Compliance with 10 CFR 50.34(B), GDC 15, GDC 21, GDC 29, and the Licensing Basis

Theresa,

You may put my initials on the SUNSI review completed box.

Best

Rich

Richard P. Correia, P.E.
Director, Division of Risk Analysis
Office of Nuclear Regulatory Research
U.S. NRC

From: Clark, Theresa
Sent: Wednesday, September 14, 2016 7:06 PM
To: Correia, Richard <Richard.Correia@nrc.gov>
Cc: Coyne, Kevin <Kevin.Coyne@nrc.gov>

Subject: Re: Memorandum From: V. McCree to G. Holahan re: Charter for Backfit Appeal Review Panel Associated with Byron and Braidwood Compliance with 10 CFR 50.34(B), GDC 15, GDC 21, GDC 29, and the Licensing Basis

Hi all--as you said would be ok in the email below, we intend to make the RES document publicly available in ADAMS at the same time as the EDO's decision documents, likely tomorrow or Friday. For ADAMS purposes, I think we will the originating office to say a SUNSI review was done. I don't think there is anything sensitive, but could I have confirmation so I can put someone's initials in the box? Thanks so much and thanks again for the report!

Theresa

On: 23 August 2016 12:25, "Correia, Richard" <Richard.Correia@nrc.gov> wrote:

Thanks Theresa for the compliments on the risk analysis report.
No objection to making it publicly available.

Rich

Richard P. Correia, P.E.
Director, Division of Risk Analysis
Office of Nuclear Regulatory Research
U.S. NRC

From: Clark, Theresa
Sent: Tuesday, August 23, 2016 12:16 PM
To: Correia, Richard <Richard.Correia@nrc.gov>
Cc: Coyne, Kevin <Kevin.Coyne@nrc.gov>; Holahan, Gary <Gary.Holahan@nrc.gov>
Subject: Re: Memorandum From: V. McCree to G. Holahan re: Charter for Backfit Appeal Review Panel Associated with Byron and Braidwood Compliance with 10 CFR 50.34(B), GDC 15, GDC 21, GDC 29, and the Licensing Basis

Rich,

Thanks again for this report. It was very helpful to the panel. As it is referenced in the panel report, we discussed today that we would like to make it publicly available at the same time as our report, if the EDO determines our report should be publicly available (i.e., not yet as it is predecisional).

Would you have any objection to it being made public at that point?

Thanks,
Theresa

On: 11 August 2016 17:54, "Correia, Richard" <Richard.Correia@nrc.gov> wrote:
Gary et al.,

The attached risk assessment report addresses the Byron/Braidwood backit issue. The conclusion is that the maximum benefit from a potential backfit remedy would provide a very small reduction in risk (i.e., less than 1E-06/year). It should be noted that the analysis contained in the report was narrowly focused on the backfit question under review by the Appeal Review Board and is intended to provide additional context and insights to the Board. As such, other applications of this information may not be appropriate unless this limitation is recognized.

Please let me know if you need any additional information or if you would like a briefing.

Regards

Rich

Richard P. Correia, P.E.
Director, Division of Risk Analysis
Office of Nuclear Regulatory Research
U.S. NRC

From: Scarbrough, Thomas
Sent: Wednesday, July 06, 2016 1:23 PM
To: Clark, Theresa
Subject: RE: Document Search
Attachments: ADAMS Document Request (T. Scarbrough) 7-6-2016.docx

Theresa,

I submitted a Form 499 today with the attached documents requested for next week.

Thanks.
Tom

From: Clark, Theresa
Sent: Thursday, June 30, 2016 3:33 PM
To: SCARBROUGH, THOMAS G <Thomas.Scarbrough@nrc.gov>
Cc: Holahan, Gary <Gary.Holahan@nrc.gov>; SPENCER, MICHAEL A <Michael.Spencer@nrc.gov>; West, Steven <Steven.West@nrc.gov>
Subject: RE: Document Search

Update—I spent some time poking through microfiche (again, one without a working printer, alas) and took some very brief notes on the contents of the various documents you referenced. See attached. I think you'll definitely want to request the starred ones to be printed. If the email you already sent ADAMSIM will do that, great. Otherwise, I'm happy to request via Form 499. Thanks!

From: CLARK, THERESA V
Sent: Thursday, June 30, 2016 1:41 PM
To: SCARBROUGH, THOMAS G <Thomas.Scarbrough@nrc.gov>
Cc: HOLAHAN, GARY M <Gary.Holahan@nrc.gov>; SPENCER, MICHAEL A <Michael.Spencer@nrc.gov>; WEST, Steven S <Steven.West@nrc.gov>
Subject: RE: Document Search

Tom—I found microfiche addresses for all of these but the EPRI Volume 6 report. See attached. I'm headed over to TWFN anyway shortly and will see if I can get any of the shorter ones to print (the machines weren't working the other day).

--
Theresa Valentine Clark
Executive Technical Assistant (Reactors)
U.S. Nuclear Regulatory Commission
Theresa.Clark@nrc.gov | 301-415-4048 | O-16E22

From: SCARBROUGH, THOMAS G
Sent: Thursday, June 30, 2016 11:25 AM
To: CLARK, THERESA V <Theresa.Clark@nrc.gov>
Subject: Document Search

Theresa,

Attached is my current list of search documents.

Thanks.
Tom

T. Scarbrough
July 6, 2016

FORM 499 DOCUMENT REQUEST

1. Applicant letter for the Braidwood nuclear power plant (Docket Number 05000456 and 457) dated July 27, 1982, from O. Kingsley (Alabama Power) to S. Chilk (NRC)
LL Accession No. 8208190307; microfiche 14387:189-14387:301

2. Electric Power Research Institute (EPRI) report NP-2628-SR, dated December 1982
LL Accession No. 8407130197; microfiche 25588:082-25588:262

3. Byron applicant (Docket Number 05000454 and 455) letter dated October 26, 1982
LL Accession No. 8211020633; microfiche 15886:171-15886:231

4. NRC letter from L. Olshan to H. Bliss, dated August 18, 1988, Subject: NUREG-0737, Item II.D.1, Performance Testing on Relief and Safety Valves for Byron Station, Units 1 and 2.
LL Accession No. 8808260355; microfiche 46653:240-46653:269

5. NRC letter from S. Sands to T. Kovach, dated May 21, 1990, Subject: NUREG-0737, Item II.D.1, "Performance Testing on Relief and Safety Valves for Braidwood Station, Units 1 and 2.
LL Accession No. 9005290209; microfiche 53927:301-53927:330

6. Letter from D. Hoffman (Consumers Power) to H. Denton (NRC), dated April 1, 1982
LL Accession No. 8207160337; microfiche 13866:001-13869:106

7. EPRI report NP-2770-LD (specifically, Volume 6). Not found during Legacy Library search by ADAMS IM team.

From: Clark, Theresa
Sent: Friday, September 09, 2016 4:39 PM
To: McCree, Victor; Johnson, Michael
Cc: Holahan, Gary; Lewis, Robert; Tracy, Glenn
Subject: RE: backfit documents

Update re: the first bullet below (Gary is already aware) – I spoke with Margie and she was happy to work with her staff to review the letter to Exelon for a quick-turnaround NLO, given that folks have already seen the panel's report. The Exelon letter is already with OGC and we expect their review by Monday.

From: Clark, Theresa
Sent: Friday, September 09, 2016 10:49 AM
To: McCree, Victor <Victor.McCree@nrc.gov>; Johnson, Michael <Michael.Johnson@nrc.gov>
Cc: Holahan, Gary <Gary.Holahan@nrc.gov>; Lewis, Robert <Robert.Lewis@nrc.gov>; Tracy, Glenn <Glenn.Tracy@nrc.gov>
Subject: backfit documents
Importance: High

Vic,

Per your request, attached are the current Word versions of the three documents we are working related to the backfit decision. ADAMS links are also provided below for completeness. I would be happy to assist in incorporating your changes into the paper/ADAMS packages. Feel free to email or call me at 301-415-4048 (office) or (b)(6) (cell).

Please note that I am also working three related matters (on which you may have views):

- Need for NLO on the letter to Exelon – per chat with Gary, I have a question in to Margie (who is in a meeting currently) to get her view
- Whether the memo to NRR should be public – my recommendation is yes, since there were panel recommendations regarding NRR activities and there could be questions
- Coordination of a press release (if we do one) as there are several process steps needed if it were to be released concurrently

Mike—FYI, as you concurred in an earlier version of the letter to Exelon.

Each of these records is publicly available in ADAMS under the specified accession number.

- **Letter responding to Exelon**
  View ADAMS P8 Properties ML16243A067
  Open ADAMS P8 Document (09/XX/16 Letter to Exelon from Victor McCree)
- **Letter responding to NEI (which had sent a letter in support of Exelon)**
  View ADAMS P8 Properties ML16246A150
  Open ADAMS P8 Document (09/XX/16 NEI Comments in Support of Exelon Generation Company Second Level Appeal (To: Anthony Pietrangelo, From: Victor McCree))
- **Memo to NRR**
  View ADAMS P8 Properties ML16246A247
  Open ADAMS P8 Document (Appeal of Backfit Imposed in Braidwood and Byron Stations (To: William Dean, From: Victor McCree))

Thanks!

--
Theresa Valentine Clark
Executive Technical Assistant (Reactors)
U.S. Nuclear Regulatory Commission
Theresa.Clark@nrc.gov | 301-415-4048 | O-16E22

From:	Clark, Theresa
Sent:	Thursday, September 15, 2016 9:56 AM
To:	'bradley.fewell@exeloncorp.com'
Subject:	FYI: courtesy copy of letter from Victor McCree
Attachments:	Courtesy Copy of Letter from VMMcCree to JBFewell - 9-15-16.pdf

The attachment is publicly avaiable in ADAMS as ML16243A067.

Good morning, Mr. Fewell,

On behalf of Victor McCree, I am sending the attached courtesy copy of his letter to you regarding your appeal of the backfit imposed on Braidwood and Byron. We expect the document to be formally dispatched, made publicly available in ADAMS, and Listserved later today.

--

Theresa Valentine Clark
Executive Technical Assistant (Reactors)
U.S. Nuclear Regulatory Commission
Theresa.Clark@nrc.gov | 301-415-4048 | O-16E22

From: Clark, Theresa
Sent: Thursday, September 15, 2016 3:40 PM
To: PIETRANGELO, Tony
Subject: courtesy copy of letter from Victor McCree
Attachments: Courtesy Copy of Letter from VMMcCree to APietrangelo - 9-15-16.pdf

Importance: High

The attachment is publicly avaiable in ADAMS as ML16243A067.

Good afternoon!

On behalf of Victor, I am sending the attached courtesy copy of his letter to you, responding to your letter in support of Exelon's appeal of the backfit imposed on Braidwood and Byron. We expect the document to be formally dispatched and made publicly available in ADAMS (along with other referenced documents) later today.

If you could please acknowledge receipt, I would appreciate it. Thanks!

--

Theresa Valentine Clark
Executive Technical Assistant (Reactors)
U.S. Nuclear Regulatory Commission
Theresa.Clark@nrc.gov | 301-415-4048 | O-16E22

From: Clark, Theresa
Sent: Friday, July 29, 2016 3:30 PM
To: Holahan, Gary
Subject: FYI: formatted references

FYI... I didn't get super far in the reference list. See S:\Backfit-Appeal\References\Formatted Reference List.docx. I can pull together all of the most important ones by the time you plan to share with NRR on Tuesday. I still think it is useful to show that you guys looked at a lot of stuff.

--
Theresa Valentine Clark
Executive Technical Assistant (Reactors)
U.S. Nuclear Regulatory Commission
Theresa.Clark@nrc.gov | 301-415-4048 | O-16E22

From: Clark, Theresa
Sent: Monday, August 08, 2016 2:41 PM
To: Scarbrough, Thomas
Cc: Holahan, Gary; West, Steven; Spencer, Michael
Subject: RE: Things to review

Thanks Tom—good comments. The ones on the references will be overcome by later changes, but the others can be merged in with what the other folks are doing when we create a master copy.

For those currently fiddling with the file—I noticed a misspelling of "Inadverent" in the first paragraph on p.27.

From: Scarbrough, Thomas
Sent: Monday, August 08, 2016 1:29 PM
To: Clark, Theresa <Theresa.Clark@nrc.gov>
Cc: Holahan, Gary <Gary.Holahan@nrc.gov>; West, Steven <Steven.West@nrc.gov>; Spencer, Michael <Michael.Spencer@nrc.gov>
Subject: RE: Things to review

Theresa,

I like your clean version of my input.

I have only a few minor edits or changes in response to your margin comments.

Attached are my redline markups of the cover letter and main report for your information. These documents are also in the Tom- Working folder.

We can include these changes in the master copy when ready.

Thanks.
Tom

Note: The blackened out text is actually pink highlighting over the words "pink highlights".

From: Clark, Theresa
Sent: Saturday, August 06, 2016 11:27 PM
To: Holahan, Gary <Gary.Holahan@nrc.gov>; West, Steven <Steven.West@nrc.gov>; Scarbrough, Thomas <Thomas.Scarbrough@nrc.gov>; Spencer, Michael <Michael.Spencer@nrc.gov>
Subject: RE: Things to review

Gentlemen,

We're really getting there. I took the files prepared to date (memo including Michael's comments, Gary's report file, and Tom's enclosures file) and created two clean files that have been formatted, edited, and otherwise prettified and such. I added some comments in the margin where I wasn't quite sure about things. Yellow highlights are for references that will need to (eventually) be put into a single format with a single list at the end—my next big project, I think. [redacted] are inserts for the future.

- S:\Backfit-Appeal\Report\cover memo (clean as of 2016 08 06 11pm).docx
- S:\Backfit-Appeal\Report\Backfit Appeal Panel Report (clean as of 2016 08 06 11pm).docx

Just because it looks all nice doesn't mean it's done (or that my edits were necessarily correct—note that we can do a compare to the last version if needed, since I accepted all of the messy changes). But, like I said, getting there. I just set up a meeting for Tuesday since we didn't have any more on the calendar, and I can set more thereafter as needed.

Thanks!

--

Theresa Valentine Clark
Executive Technical Assistant (Reactors)
U.S. Nuclear Regulatory Commission
Theresa.Clark@nrc.gov | 301-415-4048 | O-16E22

From: Holahan, Gary
Sent: Wednesday, August 03, 2016 4:58 PM
To: West, Steven <Steven.West@nrc.gov>; Scarbrough, Thomas <Thomas.Scarbrough@nrc.gov>; Clark, Theresa <Theresa.Clark@nrc.gov>; Spencer, Michael <Michael.Spencer@nrc.gov>
Subject: Things to review

Steve,
Tom,
Michael,
Theresa,

I have taken the Preliminary Findings document and incorporated it into a "Discussion" section. I added text to make it read like a report (no changes to findings or conclusions). Please review at Reports / Backfit Appeal Report 2016 08 03 2pm.

Next I will start on the Enclosures (and the sections referring to them).

I have also drafted (first draft...) a memo to Vic presenting the report. Please review at Reports / Cover memo Backfit Appeal Panel 2016 08 03 2pm.

Gary

From: Clark, Theresa
Sent: Tuesday, August 09, 2016 4:26 PM
To: Holahan, Gary; West, Steven; Scarbrough, Thomas; Spencer, Michael
Subject: RE: Draft Report

Thanks, gentlemen.

Version control is becoming even more important.

I've created a "MASTER" version (S:\Backfit-Appeal\Report\Backfit Appeal Panel Report (MASTER).docx) that incorporates all of Gary and Tom's edits, less the ones that didn't match style guide, and made a few comments about things I changed on my own. I also added the references list that I prepared separately, and will work starting tomorrow on consistently formatting the in-text citations.

To make things easier on most of us, let's use separately named feeder copies with tracked changes for inputs from now on, and I can add them into the MASTER file. Feel free to put those in the report folder and let me know you have content to incorporate.

--

Theresa Valentine Clark
Executive Technical Assistant (Reactors)
U.S. Nuclear Regulatory Commission
Theresa.Clark@nrc.gov | 301-415-4048 | O-16E22

From: Holahan, Gary
Sent: Tuesday, August 09, 2016 2:23 PM
To: West, Steven <Steven.West@nrc.gov>; Clark, Theresa <Theresa.Clark@nrc.gov>; Scarbrough, Thomas <Thomas.Scarbrough@nrc.gov>; Spencer, Michael <Michael.Spencer@nrc.gov>
Subject: Draft Report

All,

I have added my edits to the 8/6 clean version. You can see the edits (none very significant) in the comments on the 8/6/ clean version "Backfit Appeal Report (clean as of 2016 08 06 11pm).

I also saved a new clean version as "Backfit Appeal Report (clean as of 2016 08 09 2pm)

I have read all Tom's Appendix B and C and made only a few minor edits. However, I think we need to move the Appendix C "Conclusions" to another part of the report. If something needs to be done on the ISTY issue, it belongs in the main report. In addition, we need to discuss the "should" statements in Appendix C, Conclusion items 2, 3, and 4. We need to decide what's required (and what's not)... no should.

Good topics for Thursday... any more info on the IST issue will help that discussion,

Gary

From: Clark, Theresa
Sent: Friday, August 12, 2016 4:20 PM
To: Holahan, Gary; West, Steven; Scarbrough, Thomas; Spencer, Michael
Subject: RE: New section answering question 3.5

Thank you. I added a somewhat shortened version of this to the MASTER cover memo, and a somewhat edited (but not really shortened) version to the MASTER report, as well as the added reference.

From: Holahan, Gary
Sent: Friday, August 12, 2016 1:03 PM
To: West, Steven <Steven.West@nrc.gov>; Clark, Theresa <Theresa.Clark@nrc.gov>; Scarbrough, Thomas <Thomas.Scarbrough@nrc.gov>; Spencer, Michael <Michael.Spencer@nrc.gov>
Subject: New section answering question 3.5

Panel,

See Backfit-Appeal / Report / Question 3 5 Response 2016 08 12

It uses the RES report to answer question 5.
About 48 pages condensed to about two sentences.

Gary

From: Clark, Theresa
Sent: Wednesday, August 17, 2016 5:38 PM
To: Holahan, Gary; West, Steven; Scarbrough, Thomas; Spencer, Michael
Subject: FSARs & 50.59s

To add to what I sent earlier, I think I've found the last few FSAR and 50.59 puzzle pieces (at least those that might be necessary for us). See S:\Backfit-Appeal\References\BB FSARs and 5059s.

- FSAR Section 15.5, Revision 0, 1988
- FSAR Section 15.5, Revision 6, 1996 (when IOECCS was updated for overfill – PORVs credited)
- FSAR Section 15.5, Revision 7, 1998 (not huge changes – e.g., PDP assumed inoperable, possibly since there were issues with it at the time; valve stroke time added; setpoints adjusted)
- FSAR Section 15.5, Revision 11, 2004 (changes are from Revision 9 in 2002 and reference water relief through safeties)
- FSAR Chapter 15, Revision 13, 2010 (some changes from 2004 and 2006 included)
- 1994 50.59 report on safety-related power to PORVs
- 1996 and 1997 50.59 reports (seem the same) on reanalysis of the IOECCS event – seems to be response to NSAL, coincident with FSAR Revision 6
- 1999 50.59 report on automatic PORV actuation (part of several TRM changes)

If we end up discussing any of these in the report, I will add the references.

From: Clark, Theresa
Sent: Friday, August 26, 2016 8:45 AM
To: West, Steven
Subject: Re: Byron and Braidwood Backfit Panel Analysis

Ooh! Thank you for sharing!

On: 26 August 2016 08:42, "West, Steven" <Steven.West@nrc.gov> wrote:
Looks like our report is circulating. Ken is the director of the Division of Reactor Safety in Region III. Ed Hackett gave me similar feedback during a CRGR meeting yesterday afternoon.

Steve

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

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

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

Steve

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

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

Bravo!

Ken

March 23, 2016

Wednesday

March 2016						
Su	Mo	Tu	We	Th	Fr	Sa
		1	2	3	4	5
6	7	8	9	10	11	12
13	14	15	16	17	18	19
20	21	22	23	24	25	26
27	28	29	30	31		

April 2016						
Su	Mo	Tu	We	Th	Fr	Sa
					1	2
3	4	5	6	7	8	9
10	11	12	13	14	15	16
17	18	19	20	21	22	23
24	25	26	27	28	29	30

WEDNESDAY

23

7 AM	
8	Non-Responsive Record
9	Non-Responsive Record
	Exelon Drop In; EDO-OWFN-17H14-14p; McCree, Victor
10	Non-Responsive Record
11	Non-Responsive Record
12 PM	
1	Non-Responsive Record
2	Non-Responsive Record
3	Non-Responsive Record
4	Non-Responsive Record; Non-Responsive Record
5	
6	

Notes

May 3, 2016
Tuesday

May 2016						
Su	Mo	Tu	We	Th	Fr	Sa
1	2	3	4	5	6	7
8	9	10	11	12	13	14
15	16	17	18	19	20	21
22	23	24	25	26	27	28
29	30	31				

June 2016						
Su	Mo	Tu	We	Th	Fr	Sa
			1	2	3	4
5	6	7	8	9	10	11
12	13	14	15	16	17	18
19	20	21	22	23	24	25
26	27	28	29	30		

TUESDAY

3

7 AM

8 Non-Responsive Record

9

10 Non-Responsive Record

11

12 PM

1

2

3

4

Byron/Braidwood Compliance Backfit Appeal; EDO-OWFN-17H14-14p (Bridgeline:

5

6

Notes

September 12, 2016
Monday

September 2016						
Su	Mo	Tu	We	Th	Fr	Sa
				1	2	3
4	5	6	7	8	9	10
11	12	13	14	15	16	17
18	19	20	21	22	23	24
25	26	27	28	29	30	

October 2016						
Su	Mo	Tu	We	Th	Fr	Sa
						1
2	3	4	5	6	7	8
9	10	11	12	13	14	15
16	17	18	19	20	21	22
23	24	25	26	27	28	29
30	31					

MONDAY

12

Pilgrim Site Visit (Pilgrim) - Johnson, Michael To Sep 13

7 AM	
8	Non-Responsive Record
9	Non-Responsive Record
10	Non-Responsive Record; Non-Responsive Record
11	
12 PM	
1	
2	Non-Responsive Record
3	**RESERVE**
4	
5	**HIGH PRIORITY: Exelon Backfit Appeal Decision** O-17H1 (301-415-1700) McCree, Victor
6	

Notes

September 14, 2016
Wednesday

September 2016						
Su	Mo	Tu	We	Th	Fr	Sa
				1	2	3
4	5	6	7	8	9	10
11	12	13	14	15	16	17
18	19	20	21	22	23	24
25	26	27	28	29	30	

October 2016						
Su	Mo	Tu	We	Th	Fr	Sa
						1
2	3	4	5	6	7	8
9	10	11	12	13	14	15
16	17	18	19	20	21	22
23	24	25	26	27	28	29
30	31					

WEDNESDAY

14

Non-Responsive Record

7 AM

8 Non-Responsive Record

Non-Responsive Record

9

10 Non-Responsive Record

11

12 PM

1 Non-Responsive Record

2 **Exelon Drop In;** O-17B4; Johnson, Michael

Non-Responsive Record

3 Non-Responsive Record

4 Non-Responsive Record

5 Non-Responsive Record | Non-Responsive Record

6

Notes

BEST AVAILABLE COPY

September 15, 2016

Thursday

September 2016

Su	Mo	Tu	We	Th	Fr	Sa
				1	2	3
4	5	6	7	8	9	10
11	12	13	14	15	16	17
18	19	20	21	22	23	24
25	26	27	28	29	30	

October 2016

Su	Mo	Tu	We	Th	Fr	Sa
						1
2	3	4	5	6	7	8
9	10	11	12	13	14	15
16	17	18	19	20	21	22
23	24	25	26	27	28	29
30	31					

THURSDAY

15

7 AM

8 — Non-Responsive Record

9 — Non-Responsive Record

10 — Non-Responsive Record

11 — Non-Responsive Record

Non-Responsive Record

12 PM

1 — Non-Responsive Record

READ with SCIF; 2nd Floor SCIF; McCree, Victor

2 — Non-Responsive Record

3 — **Conference call re: Byron & Braidwood Backfit Appeal**; via teleconference - BRIDGE

Non-Responsive Record

4

5 — Non-Responsive Record

Conference call re: Byron & Braidwood Backfit Appeal; Victor to call Tony Pietrangelo

6

Notes

The 5:30 appointment concerned the Byron/Braidwood Backfit Appeal

From: McCree, Victor M
Sent: Friday, June 24, 2016 5:09 PM
To: CLARK, THERESA V
Subject: RE: FYI: backfit appeal panel charters

Outstanding, thanks Theresa! You have a great weekend too.

Vic

From: CLARK, THERESA V
Sent: Friday, June 24, 2016 3:30 PM
To: McCree, Victor M <Victor.McCree@nrc.gov>
Subject: FYI: backfit appeal panel charters

Vic, here are the details on the most recent sets of charters. Based on this, my leaning would be to maintain our internal EDO-level charter non-public, non-sensitive as it is purely reflective of our internal deliberations. As you mentioned, we may take the opportunity to update the guidance after this experience.

Exelon Backfit

- 6/22/2016 EDO-level appeal charter: ML16173A311 – non-public, non-sensitive (B.1)
- 1/12/2016 NRR-level appeal charter: ML15355A081 – public (released 1/14/2016 if I am reading the notes right)

Hatch Backfit

- 12/21/2011 EDO-level appeal charter: ML11341A028 – non-public, noted sensitive in ADAMS but not marked on document (keyword is B.1 which is non-sensitive)
- I spoke with Howard Benowitz regarding the Region-level appeal (since he NLOed that, ML112730194), and his recollection was that there was no formal charter. (He was going to double-check his emails.)

Also, side note—I just found that NRR did send an acknowledgement letter to Brad Fewell on the first-level appeal (ML15351A372), and Region II acknowledged the first-level Hatch appeal (ML111741416), but I think our plan just to call is fine. There's no provision requiring an acknowledgement letter at the EDO level, and I didn't see one for Hatch.

Have a great weekend!

--
Theresa Valentine Clark
Executive Technical Assistant (Reactors)
U.S. Nuclear Regulatory Commission
Theresa.Clark@nrc.gov | 301-415-4048 | O-16E22

From:	Roberts, Ashley B
Sent:	Friday, June 24, 2016 9:31 AM
To:	Keene, Todd T
Cc:	LUND, Louise L; BANIC, MERRILEE J; STUCHELL, SHELDON D; BRIGHT, RICHARD S; VALENTINE, NICHOLEE G; CLARK, THERESA V
Subject:	RE: REQUEST: PM support on backfit appeal

Todd – Richard Bright can help you out with the CACs, if you need support. I cc'd him.

Ashley Roberts (Bettis)

From: CLARK, THERESA V
Sent: Friday, June 24, 2016 9:12 AM
To: Keene, Todd T <Todd.Keene@nrc.gov>
Cc: GAVRILAS, MIRELA <Mirela.Gavrilas@nrc.gov>; LUND, Louise L <Louise.Lund@nrc.gov>; HOLAHAN, GARY M <Gary.Holahan@nrc.gov>; BANIC, MERRILEE J <Merrilee.Banic@nrc.gov>; STUCHELL, SHELDON D <Sheldon.Stuchell@nrc.gov>; Roberts, Ashley B <Ashley.RobertsBettis@nrc.gov>
Subject: REQUEST: PM support on backfit appeal

Todd,

You'd reached out previously to say you could provide PM support on the EDO appeal for the Byron/Braidwood backfit. In general, I think we have things covered up here. However, there are a few PM-type items that I could use some help on:

- **CAC for staff on the panel.** For the first appeal, there were 4 non-fee-billable CACs associated with the Byron/Braidwood dockets (MF7231-MF7324), titled "Review of Appeal of Imposition of Backfit Regarding a Condition II Event that Could Cause a More Serious Event." Could we have non-fee-billable CACs opened for the appeal as well? If it is possible to have one CAC rather than 4, that would be fine, but this may be an standard requirement. I'm cc-ing Ashley in case this is more of a PMDA item. We could also use an EDO CAC but I wasn't sure if NRR would want to have us use the dockets and such.

- **Support in scheduling a public meeting.** We don't yet have the date yet, but you're better versed in the public meeting systems than I am, so if we could have your help once Gary et al. know what they need, that would be wonderful. It's likely it will be the first week of August.

If anything else comes up, I'll let you know. Thanks much for your help!

--
Theresa Valentine Clark
Executive Technical Assistant (Reactors)
U.S. Nuclear Regulatory Commission
Theresa.Clark@nrc.gov | 301-415-4048 | O-16E22

From: Williamson, Edward
Sent: Tuesday, June 21, 2016 4:44 PM
To: Clark, Theresa
Cc: Holahan, Gary; Spencer, Michael; Lewis, Robert; Doane, Margaret; Moulding, Patrick
Subject: OGC Attorney for the Backfit Appeal Review Panel Associated with Byron and Braidwood Compliance w 50.34(b), GDC 15, 21, 29 and Licensing Basis

Hi Theresa,

The OGC Attorney for the proposed EDO's Charter Memo on the subject Backfit Appeal Panel is Mr. **Michel A. Spencer**. Mr. Spencer is a Senior Attorney in the OGC New Reactors Program Division.

Ed

Edward L. Williamson
Assosciate General Counsel
for Hearings, Enforcement and Administration
Office of the General Counsel
U.S. Nuclear Regulatory Commission

Phone (301) 415-1740