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Comment On: NRC-2015-0167-0001
Anticipated Transients That Could Develop Into More Serious Events; Request for Comment on Draft Revision to Regulatory Issue Summary

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Submitter Information

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General Comment

Please see attached file LTR-NRC-15-71.

Attachments

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LTR-NRC-15-71

September 14, 2015

Subject: **Transmittal of Westinghouse Electric Company Comments on Draft Revision 1 to Regulatory Issue Summary 2005-29 [Docket ID NRC-2015-0167]**

Dear Ms. Bladey,

Thank you for the opportunity to provide comments on the draft Revision 1 of Regulatory Issue Summary (RIS) 2005-29, "Anticipated Transients That Could Develop Into More Serious Events."

Please find enclosed the Westinghouse Electric Company (Westinghouse) comments on the draft RIS 2005-29.

For technical questions regarding the enclosed comments, please contact Andrew Detar at (412) 374-6756 or Glenn Heberle at (412) 374-5773.

James A. Gresham, Manager
Regulatory Compliance

Attachment

Westinghouse Comments on Draft Revision 1 to RIS 2005-29

Introductory Comment

The U.S. Nuclear Regulatory Commission (NRC) issued a draft revision of RIS 2005-29 seeking public comment. The purpose of this attachment is to provide comments on the draft revision on behalf of Westinghouse.

The NRC states that the intent of the revision to RIS 2005-29 (Reference 1) is "to inform addressees of concerns identified during recent license amendment reviews." Specifically, the NRC notes that the licensing bases recently reviewed "failed to demonstrate that anticipated operational occurrences (AOOs, also Condition II events) would not progress to more serious events (Condition III or IV events)." As stated in Chapter 15 of the Standard Review Plan (SRP, Reference 2), "by itself, a Condition II incident cannot generate a more serious incident of the Condition III or Condition IV category without other incidents occurring independently or result in a consequential loss of function of the RCS or reactor containment barriers." This criterion is also known as the non-escalation criterion within the context of this RIS. While draft Revision 1 of the RIS claims to "not present a new or changed NRC staff position or guidance with respect to progression of anticipated operational occurrences," it could be interpreted to set forth new requirements to demonstrate that the non-escalation criterion is met. As such, the following comments are provided in an effort to better understand the basis of the concerns outlined in the draft RIS to ensure that those licensees relying directly, or indirectly, on Westinghouse-based analysis methodology or guidance can fully comply with the intent of the non-escalation criterion. Westinghouse is available to further discuss these comments with the NRC as required.

General Comments:

1. Revision 0 of the RIS (Reference 3) acknowledged that licensees can meet the non-escalation criterion by crediting pressurizer power-operated relief valves (PORVs) and pressurizer safety valves (PSVs) that are qualified as "safety-related systems capable of relieving water." However, draft Revision 1 makes no such distinction. Removal of this distinction, which was one of the fundamental arguments on which Revision 0 was based, creates confusion about the ways that licensees can demonstrate compliance with the RIS. As such, it is requested that the differentiation between qualified and non-qualified PORVs be reinstated in Revision 1, or the basis for this change explained.

2. The addressees of the RIS are limited to pressurized water reactor (PWR) licensees. It would seem that boiling water reactor (BWR) licensees may also be susceptible to "anticipated transients that could develop into more serious events." A more thorough justification of the RIS applicability should be included to allow licensees to better determine their level of impact.

3. While Revision 0 of the RIS did not reference the General Design Criteria (GDC), draft Revision 1 states that the GDC applicable to the AOOs discussed therein (i.e., Inadvertent Operation of the Emergency Core Cooling System (IOECCS), Chemical and Volume Control System (CVCS) Malfunction, and Inadvertent Opening of a Pressurizer Pressure Relief Valve) are GDC 15, 21, and 29. This is inconsistent with the GDC identified in the SRP (Reference 2)

for these events, which are GDC 10, 13, 15, and 26 (see SRP Sections 15.5.1-15.5.2 and 15.6.1). It is unclear why the GDC have been added to the RIS and how the content of the RIS is directly related to the GDC identified. Please expand the GDC discussion, including explanation of how the applicable GDC were selected and how they relate to the RIS position.

4. Page 3 of 9 of the draft RIS (Reference 1) states the following:

> "If water relief through the PORVs continues long enough, there is potential for the rupture disc on the pressurizer relief tank to rupture and RCS water to be spilled directly into containment, leaving the plant in a condition that will require significant clean-up efforts and potentially cause dose concerns."

As written, a rupture of the rupture disc on the pressurizer relief tank could be interpreted as a violation of the Condition II criteria based on the context of the paragraph that includes this statement. While it is true that significant clean-up efforts may be required following such an event, it does not immediately violate the Condition II criteria. In fact, as long as the doses comply with applicable dose criteria (e.g., as outlined in 10 CFR 20), the rupture of the rupture disc is solely a plant economic issue and not one of nuclear safety. That is not to say that the discussion presented does not warrant consideration; however, that consideration should be based on the cost of the clean-up effort. The weight of these risks and consequences in this particular case should be undertaken by the licensee and not through regulations.

Section A Comments:

A1. Westinghouse agrees that the two CVCS Malfunction scenarios (i.e., mass addition and reactivity anomaly) are not equivalent and not interchangeable. Westinghouse also agrees that if the non-escalation criterion is being met by demonstrating that the pressurizer does not fill, it would be difficult to inherently conclude that the CVCS Malfunction event is bounded by the IOECCS event. However, the definitive conclusion that direct comparison between the CVCS Malfunction and IOECCS events is impossible does not account for plant-specific nuances. For example, if qualified PORVs are credited within the analyses, a technically valid argument could be made that since the IOECCS event fills the pressurizer faster than the CVCS Malfunction event, it is more limiting from a non-escalation perspective (since it may be a greater challenge to the PORV qualification constraints). In this case, it could be possible to demonstrate that the IOECCS event bounds the CVCS Malfunction event. Thus, it is suggested that the wording in Section A be modified to allow plant-specific approaches to be used.

Section B Comments:

B1. On Page 5 of 9 of the draft RIS (Reference 1), the Section B (IOECCS) introductory paragraph states the following:

> "... If one or more PORVs open while the pressurizer is water-solid, then the PORV(s) are generally assumed to fail open, since valves that are not qualified to relieve water are conservatively assumed to remain in the fully open position. In this case, a stuck-open

PORV could cause a Condition III SBLOCA event and fail to meet the non-escalation criterion. ..."

Westinghouse does not agree with the assertion that PORVs that open while the pressurizer is water-solid should necessarily be assumed to fail open as a consequence of relieving liquid water. The PSVs are generally not designed for sustained subcooled water discharge, which may result in failure of the valve to reseat properly, causing an unisolatable leak from the RCS pressure boundary. However, the PORVs are designed to be capable of such water relief. Furthermore, in the event that an individual PORV were to fail open or leak, the associated block valve is capable of isolating the PORV such that the integrity of the RCS pressure boundary is maintained. Thus, Nuclear Safety Advisory Letter (NSAL) 93-013 and its supplement (Reference 4) suggested that water relief via the PORVs may be credited to preclude liquid water discharge through the PSVs.

The NSALs (Reference 4) did not specifically discuss whether or not the PORVs credited to mitigate an IOECCS event should be safety-grade, or "qualified" for liquid water relief. It is understood that this is now the NRC viewpoint, as expressed in licensing review activities and the original RIS (Reference 3). However, it is important to note that this was not the case when the NSALs were issued, as outlined in the following.

In 1989, the NRC published NUREG-1316 (Reference 5) to evaluate the role of PORVs in accident management and mitigation. This was a comprehensive and detailed study that resulted in recommendations for improving the reliability of existing non-safety-grade PORVs and block valves. The findings of the report concluded the following:

> "For operating plants and construction permit holders the staff concludes it is not cost-effective to replace (back-fit) existing non-safety-grade PORVs and block valves (and associated control systems) with PORVs and block valves that are safety-grade for the sole purpose of making them safety grade when they have been determined to perform any of the safety-related functions discussed in Section 2.1 of this report or to perform any other safety-related function that may be identified in the future. Subsequent to the TMI-2 accident, a number of improvements were required of PORVs, such as requirements to be powered from Class 1E buses and to have valve position indication in the control room. Therefore, additional improvements that would result from upgrading PORVs to fully safety-grade status are considered to be of marginal benefit. ..."

In 1998, the NRC evaluated and approved the use of non-safety-grade PORVs for the mitigation of IOECCS as analyzed for a plant uprating licensing submittal. Section 3.2.12 of the Safety Evaluation Report (SER, Reference 6) stated the following:

> " ... to prevent the water solid relief from the primary code safety relief valves and the potential for the valve sticking open creating a more severe transient, [the licensee] credits the opening of the PORVs. The staff has reviewed this and determined that this is acceptable because there is sufficient time for the operators to take action and open the PORV block valve if it is closed. Additionally, although the automatic actuation of the PORV is not considered safety-related, the accumulation circuits are routed separately; there are two separate Class 1E procured transmitters, powered from 1E power supplies, with 1E relays. Therefore, the PORV is considered highly reliable and its use is acceptable."

Westinghouse is aware that the NRC adopted a different viewpoint thereafter. Plant licensees have submitted license amendment requests for which the NRC has imposed the requirement that PORVs must be fully safety-grade to be credited for liquid water relief in the analysis of pressurizer overfill events. This position appears to be based solely on the opinion of the NRC reviewer, which then became general policy, as outlined in the original RIS (Reference 3). (Note that there was no opportunity for industry comment before formal issuance of the original RIS.) It does not appear that this position has been justified by any cost-benefit analysis as required by the Backfit Rule. In contrast, the past NRC findings (Reference 5) that have since been ignored were properly supported by rigorous evaluation and a cost-benefit analysis. As a result, plant licensees submitting new analyses have been compelled to make changes to their PORVs, as necessary, to comply with this position, regardless of the cost and in the absence of any demonstrated significant benefit to plant safety.

Based on the foregoing it is clear that when the NSALs were written in 1993-1994, there was no general assumption to the effect that PORVs must be assumed to fail as a consequence of discharging liquid water. Nor was it required that PORVs must be fully safety-grade in order to be credited for safety-related functions. The draft RIS revision states (see pages 3 and 4 of 9) that several licensees have "encountered difficulties" in meeting the non-escalation criterion as a result of relying on the Westinghouse NSAL guidance. It is not clear to Westinghouse why the NRC focuses attention on the Westinghouse guidance as the source of these perceived "difficulties." Westinghouse is not aware of any NRC communications that repudiate the findings of NUREG-1316, or explicitly caution licensees about relying on the precedence of their own past licensing approvals. If the findings of NUREG-1316 are no longer acceptable, the reasons should be justified and documented.

B2. On Page 6 of 9 of the draft RIS (Reference 1), Section B.1 states the following:

> "The NRC staff noted in the original version of RIS 2005-29 that closing a block valve to isolate a stuck-open PORV is an action that would be taken to respond to a Condition III loss of coolant accident, not to a Condition II IOECCS. This demonstrates that the Condition II IOECCS does in fact become a Condition III event because a stuck open PORV is expected as the result of the event and not as an independent failure. Therefore, this approach does not demonstrate that the Condition II IOECCS meets the non-escalation criterion."

It is not clearly stated here, but based on the context of the original RIS statements referred to in this section, it is assumed that this text refers to PORVs and their automatic control system circuitry that are not qualified. If so, that should be made clear.

As discussed in the previous comment (Comment B1), Westinghouse does not agree with the assertion that PORVs that open while the pressurizer is water-solid should necessarily be assumed to fail open as a consequence of relieving water. Regardless, it must be noted that if PORVs are relied upon for mitigation of this event then the possibility of a single independent failure of a PORV (either to open or to close) must be considered, regardless of whether the PORV is safety-grade or not. The appropriate action in response to a stuck-open PORV would be to close the associated block valve, and thereafter rely upon a second functioning PORV to provide the water relief. But the NRC has stated here that closing a block valve is an action in response to a Condition III small break loss of coolant accident (SBLOCA), not a Condition II

IOECCS. Is the NRC saying that it is only an action in response to a Condition III event if the PORVs are not qualified, but it is an acceptable action in response to a Condition II event if the PORVs are qualified? How can the same mitigating action be used to define the nature of the condition that requires that action to be taken? Westinghouse believes that it cannot, because a stuck open PORV by itself does not constitute a SBLOCA, as discussed in the following.

A SBLOCA is an unisolatable leak large enough such that the charging system cannot keep up with the leak. The standard protection design for many Westinghouse plants sends a safety grade signal to close the PORVs and the associated block valves as soon as the pressure has dropped by approximately 50 psi below the nominal RCS pressure. This feature was specifically included to ensure that a Condition II event, such as a stuck open PORV, does not progress into a more serious plant condition. Furthermore, the operator could initiate a manual signal to close the valve or associated block valve as a backup to the automatic signal. There is no unisolatable break. Therefore, Westinghouse does not agree with the RIS position that as soon as liquid is relieved out of the PORVs the event has escalated into a Condition III SBLOCA event.

B3. Section B.2 of the draft RIS (Reference 1) states the following:

> "Licensing basis analyses that are based upon the application of water-qualified PSVs to mitigate IOECCS require the PSVs to open, relieve water, and reseat. The rationale relies upon the premise that none of the PORVs will open, or if they open and stay open, then they can be closed or isolated citing the same flawed logic as approach B.1 above."

As the NSALs (Reference 4) state, the general assumption was that the PSVs are not able to successfully reseat following sustained subcooled water relief. However, when the issue was first identified and the NSALs were written Westinghouse allowed for the possibility that a licensee may be able to justify that the PSVs are capable of reseating following water relief for the IOECCS event. If so, that would mean that the PORVs need not be credited to relieve water, but it also would not preclude them operating. Westinghouse does not believe that stating the fact that a stuck open PORV can be isolated by closing the associated block valve is "flawed logic" or that taking this reasonable and appropriate action in and of itself violates the non-escalation criterion. Indeed, we take exception to the NRC's logic, as explained in the previous comments.

This section of the draft RIS revision goes on to state the following:

> " ... The rationale for this licensing basis analysis may also assume that the PORVs are not operable, which the NRC staff has identified as a non-conservative assumption. It is conservative to maximize the rate at which the pressurizer fills during an IOECCS. This is done by assuming that the pressurizer PORVs and sprays are operable since they tend to limit the rate of RCS pressurization, which would permit a relatively higher rate of ECCS delivery. Thus, the pressurizer fills more rapidly as steam is relieved through the PORVs."

Westinghouse agrees that assuming operation of the PORVs to relieve steam, along with the other pressure control features, minimizes the time to fill the pressurizer. This is an important consideration for IOECCS analyses that credit PORV relief for mitigation, as the analysis defines the minimum time by which a PORV must be made available, and for some plants with limited PORV cycling capacity, the required duration of PORV operation. Therefore, for a hypothetical plant with PSVs qualified for water relief, similar relief via PORVs would still have to be evaluated for the effects on the downstream piping loads. Nowhere did the NSALs state that this would not be required.

The second paragraph of Section B.2 in the draft RIS discusses the use of PSVs for Condition II events. The following is stated:

> "... Condition II events are normally handled by the automatic pressure control system (e.g., during load rejections). If the plant conditions exceed the capabilities of the automatic pressure control system during a Condition II event, then the event should be ended with, at most, a reactor shutdown, as specified in the design requirements for Condition II events (AOOs). In contrast, PSVs will not open until after the reactor has been tripped, since the opening setpressure for the PSVs is higher than the high pressure reactor trip setpressure. In other words, PSVs are not expected to open during Condition II events."

These statements are not accurate. The design requirement statement in ANS 18.2-1973 about a Condition II event being accommodated (not "ended") with, at most, a reactor shutdown, refers to the ability to return to operation following corrective action, in contrast with the more serious events for which there may be fuel damage preventing the ability to resume operation for a considerable time, or more significant radioactive release. This requirement is not meant to imply that PSVs should not be actuated for any Condition II event, nor is that in fact the case. The loss of external electrical load/turbine trip (a Condition II event) is analyzed assuming operation of the PSVs to demonstrate that the RCS pressure limit is met. In fact, this event is the basis for sizing of the PSVs. The PORVs and other pressure control features are conservatively assumed for a separate departure from nucleate boiling (DNB) analysis case for loss of external electrical load/turbine trip, but no credit is taken in the safety analysis for overpressure protection by means of these control systems.

B4. Westinghouse generally agrees with the statements made in Section B.3 of the draft RIS (Reference 1). However, Westinghouse objects to this item being identified (at the end of the introductory paragraph for Section B) as one of the "alternative approaches suggested by NSAL-93-013." Neither the NSAL nor its supplement (Reference 4) made any reference to such an approach, nor is it considered a means of concluding that the non-escalation criterion is met.

It is noted that although this event is often conservatively analyzed for DNB as the opening of a PSV (higher relief capacity than a PORV), the actual failure open of a passive spring-loaded PSV is not a credible Condition II initiating event. The actual Condition II occurrence would be a stuck-open PORV.

B5. Westinghouse objects to Section B.4 of the draft RIS (Reference 1) being identified (at the end of the introductory paragraph for Section B) as one of the "alternative approaches suggested by NSAL-93-013." Neither the NSAL nor its supplement promoted the idea that the non-escalation criterion could be met by virtue of the IOECCS scenario being bounded by a SBLOCA. This point was only discussed in the NSAL supplement (Reference 4) section on "Assessment of Safety Significance," which acknowledges that the criterion may be violated. The purpose of the safety significance assessment is to determine the overall consequences and reportability under 10 CFR Part 21. Since the event would be bounded by SBLOCA it was concluded that the issue would not constitute a significant safety hazard. That is not the same as suggesting that it thereby meets the non-escalation criterion without further action.

Westinghouse considers the probability of an IOECCS event combined with both a failed PORV and associated block valve to be beyond that associated with the Condition II classification. Furthermore, this event does not result in severe consequences. If the failed PORV could not be isolated, the event essentially becomes a benign SBLOCA. The response would be very similar to the bleed-and-feed mitigating strategy of functional restoration guideline FR-H.1, "Loss of Secondary Heat Sink." The only difference is that with IOECCS the secondary heat sink is available, which means the ECCS and PORV would not have to assume the reactor heat load. In fact, if the event is left unchecked, both the three and four loop Westinghouse plant designs would come to an equilibrium pressure condition that is around 1000 to 1200 psia with the RCS full of either saturated or subcooled liquid and no core uncovery. Note that the two-loop plant design and some older three loop plants are not susceptible to water flow through the PORVs or safety valves for IOECCS since the ECCS shut-off head is below nominal system pressure.

B6. Based on discussion in Section B.5 of the draft RIS (Reference 1), it would seem that the NRC misinterprets the point of the statements quoted from the NSALs (Reference 4). Analyses of IOECCS that credit water relief via a PORV model the valve to function normally. The analysis does not assume that the PORV fails open as a consequence of relieving liquid water, as implied by the second issue the NRC identifies in this draft RIS revision section. Whether or not ECCS flow is "normal makeup" flow is irrelevant in this scenario. The argument being made by the NSALs is that when water is relieved through functioning (i.e., not failed-open) PORV(s) during an IOECCS event, there will not be a sustained net loss of RCS inventory because the water being relieved is the direct result of water being injected into the system. The system remains full of water, the core is not uncovered, core cooling is maintained, and thus the concerns associated with a LOCA event are not encountered. Furthermore, as previously discussed in Comments B1 and B2, the breach of the RCS pressure boundary created by a failed open PORV does not constitute a LOCA because it is isolatable. Should a single independent failure of a PORV occur (failure either to open or to close) that PORV can be isolated by closing the associated block valve, and thereafter a second functioning PORV would provide the water relief, such that the PSVs are not actuated. Thus, the non-escalation criterion is satisfied.

Section C Comments:

C1. Characterizing this event as failing to comply with the non-escalation criterion based on the potential water relief through a pressurizer PORV ignores the post-TMI requirements issued in 10 CFR 50.34 and NUREG-0737 (Reference 7). These documents required licensees to take actions to reduce the probability of a SBLOCA due to a stuck open PORV and if the probability could not be sufficiently reduced, to install an automatic PORV isolation system. Specifically, Action Item II.K.3.2 states:

> "Modifications to reduce the likelihood of a stuck-open PORV will be considered sufficient improvements in reactor safety if they reduce the probability of a small-break LOCA caused by a stuck-open PORV such that it is not a significant contributor to the probability of a small-break LOCA due to all causes. ...Based on the above guidance and

> clarification, each licensee should perform an analysis of the probability of a small-break LOCA caused by a stuck-open PORV or safety valve. This analysis should consider modifications which have been made since the TMI-2 accident to improve the probability. ...The results of the probability analysis should then be used to determine whether the modifications already implemented have reduced the probability of a small-break LOCA due to a stuck-open PORV or safety valve a sufficient amount to satisfy the criterion stated above, or whether the automatic PORV isolation system specified in Task Action item II.K. 3.1 is necessary."

The draft RIS assertions regarding this event are not in line with the post-TMI requirements already established, adopted by the industry, and approved by the NRC to address this scenario (i.e., SBLOCA caused by a stuck open PORV). The draft RIS fails to acknowledge that some plants have addressed this issue via automatic means and thus, no manual action would be necessary. For plants requiring manual action (i.e., those that did not require an automatic PORV isolation system), the draft RIS does not recognize the improvements in control room indications and operating procedures which, when combined, would result in a very early isolation of the faulted valve(s).

The draft RIS also incorrectly states that operators would isolate the valve after the ECCS flow is terminated. Per Westinghouse plant emergency procedures, closure of the PORVs and/or block valves occurs in E-0, "Reactor Trip or Safety Injection"; therefore, the valves will be isolated well before transitioning into any subsequent emergency procedure that would instruct the operators to isolate the ECCS.

The assertions made in this section seem to contradict the post-TMI requirements and represent a change in regulatory position that conflicts with the statement contained in the draft RIS under "Backfitting and Issue Finality." As such, further justification is requested to substantiate the concerns outlined in Section C of the draft RIS since it is unclear why the post-TMI requirements are no longer sufficient.

References

1. RIS 2005-29, Draft Revision 1, "Anticipated Transients That Could Develop Into More Serious Events," (ADAMS Accession Number ML15014A469).

2. NUREG-0800, Revision 3, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition," March 2007.

3. RIS 2005-29, Revision 0, "Anticipated Transients That Could Develop Into More Serious Events," December 2005. (ADAMS Accession Number ML051890212)

4. NSAL-93-013, Revision 0, "Inadvertent ECCS Actuation at Power," June 1993 and NSAL-93-013, Supplement 1, "Inadvertent ECCS Actuation at Power," October 1994.

5. NUREG-1316, Revision 0, "Technical Findings and Regulatory Analysis Related to Generic Issue 70, Evaluation of Power-Operated Relief Valve and Block Valve Reliability in PWR Nuclear Power Plants," December 1989.

6. ADAMS Accession Number ML012140259, "Safety Evaluation by the Office of Nuclear Reactor Regulation Related to Amendment No. 137 to Facility Operating License No. NPF-2 and Amendment No. 1129 to Facility Operating License No. NPF-8, Southern Nuclear Operating Company, Inc., et al., Joseph M. Farley Nuclear Plant Units 1 and 2, Docket Nos. 50-348 and 50-364."

7. NUREG-0737, Revision 0, "Clarification of TMI Action Plan Requirements," November 1980.