Mendiola, Doris

Subject: FW: public comments re Docket ID NRC-2015-0167

From: Samuel Miranda [mailto:sm0973@gmail.com]
Sent: Friday, September 11, 2015 6:06 PM
To: Bladey, Cindy
Subject: [External_Sender] public comments re Docket ID NRC-2015-0167

RECEIVED 2015 SEP 14 AM 11:15 RULES AND DIRECTIVES BRANCH

Cindy,

Here are my comments regarding Docket ID NRC-2015-0167.

You will receive a hard copy of these comments in the mail.

Sincerely,

Samuel Miranda

7/17/2015
80FR 42559
2

Public Comments

Docket ID NRC-2015-0167

Federal Register, Volume 80, Number 137, Page 42559 (July 17, 2015)

Re: the NRC's revision of regulatory issue summary (RIS) entitled, "Anticipated Transients that could Develop into More Serious Events", RIS 2005-29, Rev 1 {available in the NRC's Agencywide Documents Access and Management System (ADAMS) under accession no. ML15014A469}

In the interest of full disclosure, I identify myself as a retired NRC employee; with personal, working knowledge of the issues that this RIS addresses. I am submitting the following comments as a member of the public. They do not contain any classified, restricted or proprietary information.

My Comments:

Comment No. 1 – Applicability only to pressurized water reactors (PWRs)

The RIS applies only to transients and accidents that could occur in PWRs. The RIS should also consider transients and accidents that could occur in boiling water reactors (BWRs). If there are no BWR transients and accidents that need to be considered, then the RIS should mention this, and describe the rationale supporting this conclusion.

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Template = ADM – 013
E-RIDS= ADM-03
Add= A. Popova (BXP16)

Comment No. 2 – The impetus for this RIS

On the first page, the RIS states, "The U.S. Nuclear Regulatory Commission (NRC) is revising Regulatory Issue Summary (RIS) 2005-29[1] to inform addressees of concerns identified during recent license amendment reviews." On page 3, the RIS also states, "During recent NRC review activities, the NRC staff discovered that several licensees had encountered difficulties in meeting the non-escalation criterion for several UFSAR Chapter 15 mass addition events."

The NRC staff's discoveries are not recent. In 2002, the NRC staff noted that some licensees had not adequately demonstrated that their plant designs can meet the non-escalation criterion. Furthermore, the NRC staff noted that many of these licensees were applying solutions that were not consistent with federal regulations (e.g., 10 CFR 50), or their plant licensing bases, and certain of the rationales could not even be supported by the basic principles of fluid mechanics. The NRC staff learned that these unusual rationales were drawn from a nuclear safety advisory letter (NSAL) that was written by Westinghouse, in 1993, and distributed to its customers; but not to the NRC. At least one of the licensees cited this NSAL in its docketed license amendment request (LAR). Consequently, the NRC staff was able to request and receive a copy of this letter, NSAL 93-013, as part of its LAR technical review. The NRC staff identified a number of objectionable approaches, in the NSAL, that could not possibly be accepted in any of the staff's LAR reviews. The number of licensees that were applying these questionable rationales prompted the NRC staff to publish RIS 2005-29 (ADAMS No. ML051890212), in December, 2005.

Now, a revised and expanded version of RIS 2005-29 is to be published, a decade after the original. It took 12 years to publish RIS 2005-29, and another 10 years to publish RIS 2005-29, Rev 1. In other words, RIS 2005-29 and RIS 2005-29, Rev 1 address certain, problematic concepts in NSAL 93-013. Accordingly, some of the following comments will refer to NSAL 93-013, and its effect upon the revised RIS, licensing actions and NRC staff reviews.

Comment No. 3 – Responses to the RIS

On the first page, the RIS states, "Revision 1 to RIS 2005-29 supersedes in its entirety RIS 2005-29. Neither the original RIS nor this revision transmits any new requirements or requires any specific action or written response."

So far, only about half a dozen plants have taken steps to comply with the non-escalation criterion (e.g., by upgrading the quality of their pressurizer power-operated relief valves (PORVs) to safety grade, which is accepted for use in mitigating accidents). The remainder, about two dozen plant designs, do not adequately satisfy the non-escalation criterion. These plants are being operated outside their licensing bases.

The owners/operators of the few plant designs that meet the non-escalation criterion have not acted in response to RIS 2005-29. Their corrective actions re generally responses to incidents in their operating histories, or to NRC staff questions that were posed during reviews of licensees' requests for raising their plants' authorized operation power levels. RIS 2005-29, therefore, has been largely ineffective. Ten years later, the NRC staff proposes to use another (revised) RIS to communicate its concern with this issue. It seems that it would be more effective to use a more assertive vehicle, such as a Generic Letter, which can compel licensees to make individual responses.

Comment No. 5 – The meaning of "not approved for use"

Page 4 of the RIS states that NSAL 93-013 has not been "approved for use". The footnote defines "approved for use" and "endorsed"; but not the meaning of "not approved for use". The footnote states, "With respect to third-party documents, 'approved for use' means that the NRC has determined that the matters addressed in the document are technically acceptable and consistent with NRC regulatory requirements, guidance and policy, but the NRC neither supports nor discourages entities from using the positions set forth in the document." This is interesting information; but it's not relevant to the gist of the RIS.

The relevant definition, for this RIS, is the meaning of "not approved for use". This RIS specifically addresses certain rationales that are not approved for use, with the objective of ending their use by licensees. The definition of "not approved for use" is important enough to be written into the body of the RIS; not hidden away in a footnote. It should state that the NRC has determined that certain approaches, in this NSAL, are NOT technically acceptable and/or NOT consistent with NRC regulatory requirements, guidance and policy. It should also state that the NRC will NOT accept certain, specified rationales if they're used in licensees' LARs.

Comment No. 6 – The availability of NSAL 93-013

Page 4 of the RIS states that the supplement to NSAL 93-013 is available in ADAMS, under accession no. ML050320117. NSAL 93-013 was issued on June 30, 1993, and it was supplemented on October 28, 1994. Both documents are available in ADAMS, under accession no. ML052930330. The reference should be changed, to give readers access to both documents.

Comment No. 7

In June, 1993, NSAL 93-013 (ADAMS No. ML052930330) advised licensees that, "Although Westinghouse previously adopted the conservative criterion of preventing the pressurizer from becoming water solid, the acceptability of water leakage from the RCS for Inadvertent Operation of ECCS Condition II events is supported by NUREG-0800 and ANS-051.1. To meet the applicable Condition II criteria, the magnitude of any water relief must not exceed that of the normal makeup systems (which it will not by definition since this is the cause of the water relief), and the ability to orderly shutdown the reactor must be maintained."

The NSAL goes on to state, "Credit the use of one or more PORVs to help mitigate the accident. Preliminary sensitivity analyses have shown that if a water solid pressurizer condition is reached, one PORV should be sufficient to maintain pressure below the PSRV setpoints and prevent discharge of water through the pressurizer safety relief valves." Thus, the NSAL shifts analysis success criterion from preventing the pressurizer from becoming water solid to preventing the opening of a PSRV (safety valve). According to the NSAL, "Water relief through the PORVs is not a concern because the PORV block valves would be available to isolate the PORVs should they fail to close."

In its NSAL supplement of October, 1994, Westinghouse repeats this rationale: "Since the cause of the water relief is the ECCS flow, the magnitude of the leak will be less than or equivalent to that of the ECCS (i.e., operation of the ECCS maintains RCS inventory during the postulated event and establishes the magnitude of the subject leak)." This is not true during the event. It could be true later, i.e., long after the event is over, when the RCS pressure decreases to a level that allows the inflow and outflow rates to balance.

In RIS 2005-29, the NRC staff objects to the use of unqualified PORVs to relieve water. Among other things, the staff supports its objection with reference to RS-001, "Review Standard for Extended Power Uprates," of 2003, which states, "For the inadvertent operation of emergency core cooling system and chemical and volume control system malfunctions that increase reactor coolant inventory events: (a) non-safety-grade pressure-operated relief valves should not be credited for event mitigation and (b) pressurizer level should not be allowed to reach a pressurizer water-solid condition." The disagreement continues to this day. RIS 2005-29, Rev 1 is clear about the concept of critical flow in Section B.5, and repeats the NRC staff's position that the use of unqualified PORVs to relieve water is not acceptable.

Despite the NRC staff's objections a number of licensees continue to claim that the non-escalation criterion is met by showing that their PSRVs would not open. The RIS should explicitly state that this rationale will not be accepted.

There is a precedent for this position. In 1997, PSE&G claimed that operation of the PORVs in their Salem plants would prevent the PSRVs from opening and relieving water. During the technical review, the NRC staff rightly noted that the PORVs were not "safety-related" equipment. Consequently, PSE&G upgraded the PORVs to qualify them for use in accident mitigation. This was done by License Amendments 194 and 177, respectively, for Units 1 and 2 (ML011720397).

<u>Comment No. 8 – Permission to violate an "applicable licensing criterion"</u>

NSAL 93-013 states, "Without appropriate operator action to terminate safety injection flow prior to reaching a water-solid pressurizer condition, the Inadvertent ECCS Actuation at Power event may progress from a Condition II to a more severe Condition III LOCA event as described above. While this occurrence may result in a violation of one of the applicable licensing basis criteria for a Condition II event, it is not considered a significant safety concern." (LOCA means loss of coolant accident.)

The RIS should state that a licensee, may not exempt itself from meeting, "one of the applicable licensing basis criteria for a Condition II event." Exemptions are issued by the NRC, after due review; and not by Westinghouse.

Comment No. 9 – Reporting requirements

NSAL 93-013 states, "Westinghouse is unable to determine if this issue would cause a substantial safety hazard or a failure to comply resulting in a substantial safety hazard because sufficient plant specific information is not available. This information is being transferred to the applicable plants pursuant to 10 CFR 21.21(h). The NRC has not been notified of this issue." Consequently, NSAL 93-013 was disseminated to Westinghouse's customers; but not to the NRC.

Also in NSAL 93-013, Westinghouse concludes that, "this occurrence may result in a violation of one of the applicable licensing basis criteria for a Condition II event, it is not considered a significant safety concern."

Apparently, Westinghouse has "sufficient plant specific information" to determine that there is no "significant safety concern"; but not enough "plant specific information" to determine whether there is a "substantial safety hazard".

a. The NRC should ascertain the difference between "significant safety concern," and "substantial safety hazard", as used by Westinghouse in NSAL 93-013.

b. The RIS should explain how this difference could account for Westinghouse's ability to conclude that there is no "significant safety concern" in one instance; but not to determine whether there is a "substantial safety hazard" in the other instance. (This could require an identification, from Westinghouse, of the plant specific information it needs to make a determination that there is a "substantial safety hazard".)

Comment No. 10 - Competence

NSAL 93-013 (supplement) states, "Without appropriate operator action to terminate safety injection flow prior to reaching a water-solid pressurizer condition, the Inadvertent ECCS Actuation at Power event may progress from a Condition II to a more severe Condition III LOCA event as described above. While this occurrence may result in a violation of one of the applicable licensing basis criteria for a Condition II event, it is not considered a significant safety concern. As a LOCA event, discharge of coolant out of the PSRVs and PORVs due to ECCS flow is not significantly adverse relative to other Condition III LOCA events currently analyzed. This is because the pressurizer is located on the hot leg (a hot leg LOCA being less severe than a cold leg LOCA) and because the Inadvertent ECCS Actuation at Power event typically models maximum ECCS flow (to maximize the effects of the Initiating event) which is a benefit for LOCA. As such, the

Inadvertent ECCS Actuation at Power induced LOCA Is bounded by the existing small break LOCA analyses." This rationale is repeated in NSAL 07-10, "Loss-of-Normal Feedwater/Loss-of-Offsite AC Power Analysis PORV Modeling Assumptions", which was issued in November, 2007. This was 14 years after the release of NSAL 93-013, and two years after the publication of RIS 2005-29.

There could be some Condition III LOCA analyses, in the licensing bases, that yield more severe results than the results of Inadvertent ECCS Actuation at Power events that have progressed into LOCAs; but this doesn't mean that these LOCAs bound the aggravated Inadvertent ECCS Actuation at Power events. This argument neglects the difference between the two cases' frequencies of occurrence. The LOCA that begins as an Inadvertent ECCS Actuation at Power event remains a Condition II event. It must be judged against Condition II acceptance criteria; and cannot be compared to a Condition III event.

Westinghouse's persistent advancement of this sort of "evaluation" implies a longstanding failure to understand the non-escalation criterion, and why it was adopted, in 1973. Bounding analyses, in this manner, undermines the fundamental scheme of event classification, upon which all nuclear safety analyses are based. It could effectively create a new category of high-frequency; severe-consequence events.

This can be construed as a lack of competence in the world's leading supplier and designer of nuclear power plants. It can also be viewed as a safety concern:

a. When the "evaluation" goes uncorrected for 14 years;

b. When it is accepted by licensees and incorporated into their LARs, which are submitted to the NRC under oath and affirmation;

c. When it is accepted (sometimes) by NRC staff members who review the aforementioned LARs;

d. When future "evaluations" of this kind go unchallenged.

The question of competence can also be raised when licensees claim that their analyses of reactivity perturbations due to a chemical and volume control system (CVCS) malfunction (i.e., boron dilution) also serve to demonstrate compliance with the non-escalation criterion.

Comment No. 11 - No significant hazards consideration

10 CFR § 50.92 states, "(c) The Commission may make a final determination ... that a proposed amendment to an operating license for a facility licensed under § 50.21(b) or § 50.22 ... involves no significant hazards consideration, if operation of the facility in accordance with the proposed amendment would not:

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

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

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

It seems that LARs, that are proposed for application to plant designs that do not meet the non-escalation criterion cannot meet the requirements of 10 CFR § 50.92, due to the uncorrected existence of an underlying design flaw.

In plant designs that do not meet the non-escalation criterion, a Condition II event (e.g., an inadvertent operation of the ECCS) could become a Condition III or IV event (e.g., a LOCA) without the occurrence of another, independent fault. In such plants, certain Condition III or IV events can have Condition II probabilities of occurrence. Thus, the probabilities of occurrence for certain Condition III or IV events is increased. Requirement (1) is not satisfied.

Condition III or IV events that have Condition II probabilities of occurrence must be evaluated against Condition II acceptance criteria. Certain licensees, following advice from NSAL 93-013, claim that small LOCAs that result from stuck-open PORVs are already evaluated in their licensing bases. They are not. For example, small LOCAs that begin as Condition II events must be evaluated as Condition II events, i.e., to show that they will not result in any fuel damage. The absence of such (acceptable) evaluations indicates that Requirement (2) is not satisfied.

Creating the possibility of a new or different class of accidents from any accidents previously evaluated (i.e., the class of Condition III and IV events, with the relatively higher Condition II probabilities of occurrence), significantly reduces the margin of safety. This indicates that Requirement (3) is not satisfied.

The RIS should state that a licensee cannot claim that the requirements of 10 CFR § 50.92 are met, when submitting an LAR, until it has first demonstrated that the fundamental plant design, upon which the LAR is based, is in compliance with the non-escalation criterion.

The RIS should also state that the NRC cannot approve a change in licensing basis (e.g., in plant design or operation) until it verifies that the *status quo ante* is acceptable.

<u>Comment No. 11 – History and examples</u>

The RIS refers to one incident, in which the pressurizer filled, and caused water to be relieved through the PORVs (see Footnote 5, and ADAMS Accession No. ML051860338). This incident occurred at the Millstone Unit 3 plant. Millstone Unit 3 is one of the few plants that is equipped with safety grade PORVs (i.e., they are, among other things, qualified for water relief, and the plant design meets the non-escalation criterion). Perhaps the RIS should include a description of the incident, and a summary of the NRC inspectors' conclusions. The RIS could also compare and contrast the NRC inspectors' conclusions with the NRC inspectors' conclusions of another, similar incident that occurred in a plant that does not meet the non-escalation criterion.

Comment No. 12 - Backfitting

The RIS should expand the backfitting section to note that 10 CFR 50.109, "Backfitting", permits the NRC to order a modification that "is necessary to bring a facility into ... conformance with written commitments by the licensee". This would be a "compliance" backfit. To this end, the RIS should include at least one example of a backfit order, wherein the NRC staff ordered a licensee to perform the plant design modifications and/or evaluations to demonstrate compliance with the non-escalation criterion. The example(s) should specify the licensee, the affected plant(s), the date(s) of the order(s), and the corresponding ADAMS Accession number(s).

The RIS could also include a list of plants that could become candidates for backfitting.

Comment No. 13 – Three Mile Island considerations

ECCS actuations often occur as the result of other Condition II events, such as turbine trips or inadvertent openings of PORVs. These ECCS actuations are not inadvertent; but they are often unnecessary. Operators are directed to terminate the ECCS flow according to specified procedures, and within a limited time interval. The RIS alludes to this in its discussion of the inadvertent opening of a PORV event.

The RIS should examine how such Condition II events can resemble certain aspects of the Three Mile Island accident, and consider the circumstances under which a "substantial safety hazard" could be posed. For example, consider this sequence of events:

1. The circulating water pumps tripped, which led to a trip of the reactor and turbine
2. When the turbine stop valves closed, they sent a pressure wave through the main steam piping
3. The steam system pressure transient, combined with low reactor coolant temperature, caused an automatic actuation of the safety injection system
4. The safety injection flow caused the pressurizer to fill solid with water
5. The safety injection flow, when pumped a system with an already solid pressurizer, caused the PORVs to open cycle numerous times
6. The water relief, through the PORVs, eventually led to the opening of the pressurizer relief tank rupture disks.

This happened at Salem on April 7, 1994 (ADAMS ML003702822).

In 1993, NSAL 93-013 advised customers that, "Water relief through the PORVs is not a concern because the PORV block valves would be available to isolate the PORVs should they fail to close." Here is an instance wherein the safety injection actuation was not inadvertent. If the PORVs had failed open (due to the water relief), and the PORV block valves had been closed before the safety injection flow was terminated, the pressurizer pressure could have risen to the safety valve opening setpoint. Safety valves that fail open cannot be isolated. If the safety injection flow had been ended, in the presence of a failed open PORV or PSRV, and the safety injection flow could not be restored, or could not be restored in time, then the event would begin to look like the Three Mile Island accident. In any case, the opening of the pressurizer relief tank rupture disks, and the subsequent spillage of reactor coolant into the containment, complicated recovery and restart operations to a level that is beyond the definition of Condition II events.

There was another notable incident in which a PORV opened and stuck open. This occurred on August 20, 1974, at the Westinghouse-designed Beznau Unit 1 plant, in Switzerland (ADAMS ML031320181). A description of this incident is in the official records of the President's Special Commission that investigated the TMI-2 accident.

1. At 11:20 AM, there was a disturbance in the external grid which caused one of Beznau's two turbines to trip (on high casing vibration). A single turbine trip in a two-turbine plant is equivalent to a 50% load rejection, and this should not demand a reactor trip if the plant's control systems are working properly. However, the Beznau steam dump system was not working.

2. Feedwater flow, steam flow and steam generator level decreased.

3. Pressurizer pressure rose rapidly to the PORV opening setpoint in about 11 seconds. Both PORVs opened.

4. Pressurizer pressure decreased below the PORV closing setpoint; but one PORV did not shut. This resulted in a reactor trip on low pressurizer pressure, in about 49 seconds.

5. The reactor trip signal tripped the turbine which was still in operation, which caused a further increase in steam pressure, to the opening setpoint of the steam generator safety valves.

6. Reactor coolant system (RCS) average temperature and pressurizer pressure and level decreased.

7. At about one minute after the reactor trip, the pressurizer pressure had fallen to hot leg saturation. Subsequently, hot leg flashing resulted in an increase of pressurizer level until the pressurizer filled (about 3 minutes after reactor trip); probably resulting in water discharge from the open PORV, and bulk boiling in the core.

8. The operator isolated the failed PORV.

9. Pressurizer level decreased to the safety injection actuation setpoint (safety injection was actuated by coincident low pressurizer pressure and level), at about 11 minutes after reactor trip. (Note: the coincidence requirement in the safety injection actuation logic was deleted after the Three Mile Island accident. See Action Item 3 of IE Bulletin 79-06A.)

10. The system then started refilling. When pressurizer-level reached about 70%, the operator shut down the safety injection pumps.

After the incident, it was seen that the yoke, in the PORV pressurizer relief valve which had failed to close, had broken off completely. The break showed the presence of a very large flaw. The broken faces showed classic brittle failure together with evidence that the faces had rubbed together following failure. It was also possible that the valve spindle had been slightly bent.

Two US plants used the same PORV design; which was implemented with the same materials.

Comment No. 14 – Condition II events don't always occur spuriously

The events addressed in the RIS, the inadvertent ECCS actuation, and the spurious opening of a PORV, are Condition II events, which can occur during the lifetime of a plant. The non-escalation criterion does not allow a Condition II event to develop into a Condition III or IV event without the occurrence of another, independent fault. In licensing basis accident analyses, the inadvertent ECCS actuation, and the spurious opening of a PORV, like all postulated evens, are evaluated individually (i.e., without the occurrence of other Condition II events). In real life, the ECCS can be actuated, or a PORV can be opened during the predicted course of another event, such as a reactor or a turbine trip. If this occurs, the non-escalation criterion would continue to apply, since these events are not independent. That is, Condition II events that occur, as the logical result of other Condition II events are not independent faults, and they are not inadvertent actuations that are demanded by the automatic control or protection systems; but they're often unnecessary actuations. As such, they would have to be handled (e.g., terminated), probably by the operator; before they develop into Condition III or IV events.

When considered in this context, the overall frequencies of occurrence of actuations of the ECCS, and openings of one or more PORVs could increase, from once in a plant's lifetime, to several times in a year of operation. The higher frequencies still fall within the range of Condition II events; but the NRC and licensees should evaluate whether the higher frequencies can introduce questions of reduced safety margin, or even create a "significant safety hazard". For example, in the 1980s, the NRC notified licensees and vendors that the high frequencies of unnecessary reactor trips (which usually occurred during startup operations) were considered to be a safety concern, since they unnecessarily challenged safety systems.

Comment No. 14 - Accountability

Licensees have committed to meet the non-escalation criterion since it appeared in 1973 (ANS 18.2-1973, "Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants). The AEC (and then the NRC) has been issuing operating licenses based, in part, on that commitment. When some of its customers encountered difficulties in demonstrating compliance with the non-escalation criterion, Westinghouse issued NSAL 93-013 to recommend a variety of methods by which compliance could be achieved. Unfortunately, certain of Westinghouse's recommendations are not acceptable to the NRC. The NRC identified them in RIS 2005-29. Ten years later, the NRC reiterated its concern, in RIS 2005-29, Rev 1. Today, there are still more than a dozen plant designs that do not meet the non-escalation criterion.

In hindsight, it seems that NSAL 93-013 proved to be a source of confusion and controversy; not the useful reference that Westinghouse had intended. Certain aspects of NSAL 93-013 were instrumental in making it necessary for the NRC to issue a RIS, in two versions. Furthermore, Westinghouse avoided performing a 10 CFR 21 evaluation by passing the question onto its customers. Westinghouse claimed it did not have the plant specific information it required to determine whether a "substantial safety hazard" could exist in the plants that it has designed and sold to its customers.

The NRC should also recognize that NSAL 93-013 has been, and continues to be largely counterproductive in the aforementioned effort to correct the situation. The NRC also should (1) ask Westinghouse to revise or retract NSAL 93-013, and (2) to perform the 10 CFR 21 evaluation it should have performed in 1993.

Today, NSAL 93-013 continues to dispense advice that could lead licensees into undertaking risky licensing strategies. The NRC staff should give certain licensees due recognition for incorporating one or more of the unacceptable rationales, drawn from NSAL 93-013, into their LARs.

The NRC staff should train its reviewers to understand why certain approaches, advocated by Westinghouse in NSAL 93-013, are not acceptable. Then reviewers could more easily identify licensees' use of these concepts in the LARs that come under their review.

There remain more than a dozen plants, most of them of Westinghouse design that are operating outside their licensing bases. The NRC should identify them, and move to bring them into compliance with their commitments to satisfy the non-escalation criterion.