January 25, 2017

Mr. Victor M. McCree, Executive Director for Operations
U.S. Nuclear Regulatory Commission,
Washington, DC 20555–0001

SUBJECT: Enforcement Petition (10 CFR §2.206) Regarding Westinghouse Electric Corporation

Samuel Miranda (the Petitioner) hereby submits this Petition, pursuant to the terms of 10 CFR §2.206, regarding certain, erroneous advice that Westinghouse Electric Corporation (a.k.a. Toshiba) has disseminated to its customers through its series of Nuclear Safety Advisory Letters (NSALs). For example, a particularly problematic letter, NSAL-93-013 [1], was disseminated to the following US plant operators:

Braidwood 1 & 2	V. C. Summer	D. C. Cook 1 & 2	Shearon Harris
W.B. McGuire 1 &2	Catawba 1 & 2	Beaver Valley 1 & 2	J.M. Farley 1 & 2
Vogtle 1 & 2	Seabrook	Millstone 3	North Anna 1 & 2
Surry 1 & 2	Salem 1 & 2	Diablo Canyon 1 & 2	Wolf Creek
Callaway	Sequoyah 1 & 2	Watts Bar 1 & 2	

(This NSAL was also sent to about an equal number of foreign plant operators.)

The Petitioner maintains that several of Westinghouse's customers have adopted some of Westinghouse's NSAL suggestions, and inserted them into their license amendment requests (LARs) for power uprating authorizations and other licensing actions. Furthermore, these LARs were accompanied by statements of *no significant hazards*, as per 10 CFR §50.92, made under Oath and Affirmation. Westinghouse, on the other hand, did not notify the NRC staff of any safety defects, under the terms of 10 CFR §21.

The Petitioner requests the NRC to take the following actions:

(1) Publish a generic communication, e.g., a Generic Letter, to inform licensees of the problematic advice in Westinghouse's NSALs. [1] [9] Consider GL 79-45 as an example, and precedent. [14]
(2) Revise the applicable standard review plans (in NUREG-0800) to alert NRC reviewers to licensees' implementations of certain, unacceptable items of advice from Westinghouse's NSALs.
(3) Require Westinghouse to submit all of its NSALs to the NRC staff for information, and review.
(4) Require Westinghouse to make the applicable NSAL retractions, and revisions, and file the relevant 10 CFR §21 reports.
(5) Require Westinghouse to copy the NRC staff in its distribution of future NSALs.
(6) Fine Westinghouse, an amount that is sufficient to reimburse the NRC staff for the taxpayers' money it spent to review LARs that were based upon faulty advice in Westinghouse NSALs.
(7) Fine Westinghouse, an additional amount, as a penalty for this longstanding, continuing deception.

Please read Attachments (B), (C), and (D) for the particulars:

<u>Attachment</u>
(A) Summary of the Petitioner's education and experience, and disclosures,

(B) Description and evaluation of specific errors in certain Westinghouse NSALs,
(C) Safety Significance
(D) Summary, Conclusion
References

Please contact the Petitioner for additional details.

With respect and concern,
Samuel Miranda, PE
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<u>Attachment A</u>

<u>A summary of the Petitioner's education and experience in nuclear safety analysis and licensing</u>

The Petitioner, Samuel Miranda, holds Bachelor's and Master's degrees in nuclear engineering from Columbia University, and a Professional Engineer's license in mechanical engineering, in the Commonwealth of Pennsylvania.

The Petitioner has more than 40 years of experience in reactor safety analysis and licensing at Westinghouse and the NRC.

At Westinghouse (25 years), the Petitioner worked in their Nuclear Safety Department, where he performed nuclear safety analyses of Westinghouse plants, CE-designed plants, and Soviet VVER plants to obtain construction permits and operating licenses, to resolve reactor safety questions, to improve nuclear power plant operability, and to support the licensing of nuclear plant modifications, core reloads, and changes in operating procedures. He also developed standards and methods for use in nuclear safety analysis, and automatic reactor protection systems design. His work in reactor protection systems design included the preparation of functional requirements, component sizing, and determination of setpoints, time response limits, and Technical Specification revisions. In the 1980s, the Petitioner managed a program, for more than 30 utilities in the Westinghouse Owners Group, to develop a system to improve power plant availability and safety by reducing the frequency of unnecessary automatic reactor trips (see patent no. 4,832,898).

At the NRC (14 years), the Petitioner worked in NRR's Division of Safety Systems (DSS), where he reviewed license amendment requests (LARs) for license renewals, power upratings, and modifications of protection systems in PWR and BWR reactor systems. This included presenting and defending review results before the Advisory Committee on Reactor Safeguards (ACRS). He also revised several sections of the Standard Review Plan (NUREG-0800), and presented them to the ACRS. The Petitioner wrote RIS 2005-029 [12] regarding compliance with the design requirement that is the subject of this Petition.

The Petitioner retired from the NRC in August, 2014, at grade GG-15.

<u>Disclosures:</u>

(1) The Petitioner was directly involved in the production of References [2], [8], and [10].
(2) While working at Westinghouse, the Petitioner was not involved in the production of [1].
(3) Also while working at Westinghouse, the Petitioner reported, directly, to one of the authors of [5].

Attachment B

Description and evaluation of specific errors in certain Westinghouse NSALs

Attachment B will focus upon mass addition events, such as the Inadvertent Actuation of the Emergency Core Cooling System (IOECCS), and the Chemical and Volume Control System (CVCS) Malfunction, which are reported in plants' licensing bases, and addressed in Westinghouse's NSALs [1] and [9].

The General Design Criteria (GDCs) [3] define two basic types of events: anticipated operational occurrences" (AOOs), and "postulated accidents" (PAs). AOOs are, *those conditions of normal operation which are expected to occur one or more times during the life of the nuclear power unit*[1]. PAs are less frequent; but more serious events, such as loss of coolant accidents (LOCAs). If risk is defined as the product of consequences and frequency of occurrence, then the risk of an AOO would be about the same as the risk of a LOCA. This principle was established in 1971, and published in 1983 [4], *The nuclear safety criteria … have been established on the premise that: a. Those situations in the plant that are assessed as having a high frequency of occurrence shall have a small consequence to the public, and b. Those extreme situations having the potential for the greatest consequence to the public shall be those having a very low frequency of occurrence.* The IOECCS, and the CVCS Malfunction are both AOOs.

According to the American Nuclear Society (ANS) standard of 1973 [5], AOOs "… *shall be accommodated with, at most, a shutdown of the reactor with the plant capable of returning to operation after corrective action.* This requirement was acknowledged, by Westinghouse, in its NSAL of 1993 [1]. In summary, AOOs must not require anything more, for protection, than a reactor shutdown.

The following requirements pertain to the evaluation of AOOs. They are based upon the requirements in [3], [5] and [6]. There are more than a dozen AOOs.

1. Pressure in the reactor coolant and main steam systems should be maintained below 110% of the design values,

2. Fuel cladding integrity shall be maintained by ensuring that the minimum departure from nucleate boiling ratio (DNBR) remains above the DNBR limit, derived at a 95% confidence level and 95% probability, and

3. An incident of moderate frequency should not generate a more serious plant condition without other faults occurring independently.

These requirements: (1), (2), and (3), will hereinafter be known as the *overpressure requirement,* the *DNB requirement,* and the *non-escalation requirement*, respectively. The *non-escalation requirement* is particularly important, since it protects the integrity of the fundamental principle that underlies nuclear plant design and analysis, and defines the aforementioned system of categorizing events, and specifying limits for events' consequences. [5]

Certain AOOs (e.g., the IOECCS) cause an excessive amount of water to be added to the reactor coolant system (RCS) [2] [6]. This could generate a more serious plant condition by filling the pressurizer, and

[1] Quotations, and locally defined terms are denoted by the use of *italics* throughout this Petition.

raising its pressure to the power-operated relief valve (PORV) opening setpoint, and thereby causing the PORVs to open and relieve water. If the PORVs are not qualified to relieve water, then they cannot be relied upon to reseat (i.e., they must be assumed to stick open). In this way, stuck open PORVs produce a more serious plant condition (e.g., a small LOCA) by opening a hole at the top of the pressurizer. If there is no concurrent instance of another, independent fault or operator error, then the resulting LOCA is evidence that the plant design does not meet the *non-escalation requirement*. If the PORVs are not available (e.g., if the plant is operating with isolated PORVs), then the RCS pressure could rise to the pressurizer relief safety valve (PRSV) opening setpressure (set to the RCS design pressure of 2,500 psia).

This Petition will examine, in detail, the advice Westinghouse renders in [1] and [9], and present information to show how certain aspects of this advice, if applied, can be harmful to nuclear plant safety and design.

(a) PRSVs may not be used in lieu of PORVs

The NSAL [1] suggests that the IOECCS and CVCS Malfunction AOOs may be mitigated by opening the PRSVs (a.k.a. pressurizer safety valves, or PSVs), if they are qualified to relieve water. To apply the PSVs, the PORVs, which are set to open at a relatively lower pressure, must be prevented from opening. This is necessary because, if *one or more PORVs (are) available, the PSRV setpoint will not be reached.* [9] Pressure relief, either by steam or water, will be provided by the PSVs only if all the PORVs remain closed. This requires manually closing the PORV block valves, or operating the plant with isolated PORVs.

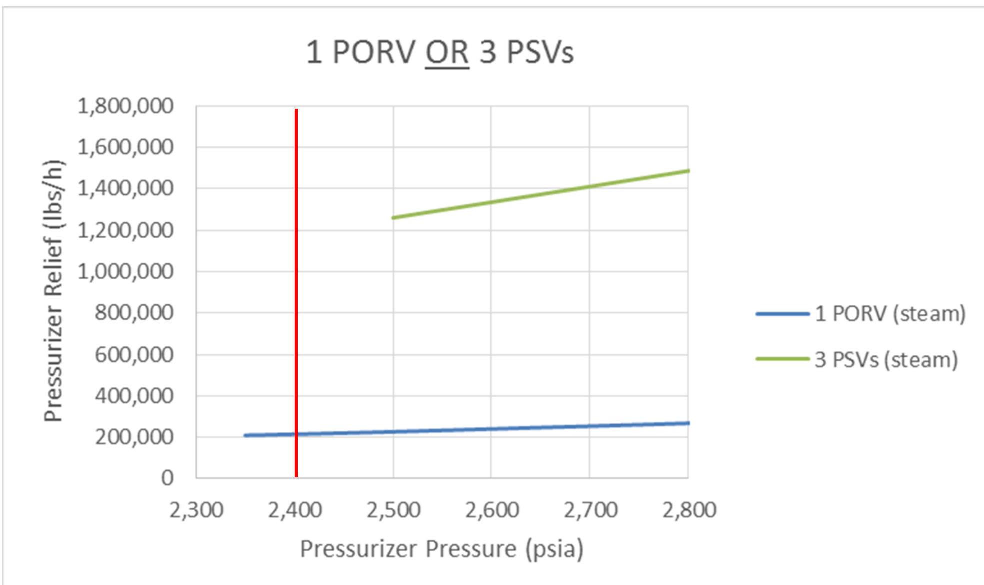
Recall that AOOs "… *shall be accommodated with, at most, a shutdown of the reactor.* [5] This can stated in logical form (if A then B) by: *If the event is an AOO, then it can be accommodated with a reactor shutdown.* If this is true, then the contrapositive (if not B then not A) must also be true*. If the event cannot be accommodated by a reactor shutdown, then it is not an AOO.* Therefore, events that pressurize the RCS past the reactor trip line (in the following figure) are not AOOs.

The following table shows that PORVs are intended for use during AOOs and PSVs are intended for use in events that are more serious than AOOs.

	PORVs	PRSVs or PSVs
Description	2 or 3 pilot-operated relief valves (part of the pressurizer pressure control system, along with heaters and spray)	3 spring-loaded safety valves
Quality	Control grade	Safety grade
Typical Capacity, per valve	210,000 lbs/hr (steam) at 2,350 psia	420,000 lbs/hr (steam) at 2,500 psia
Purpose, and Safety Function	Prevent unnecessary reactor trips (at 2400 psia), and challenges to PRSVs (at 2500 psia)	Prevent overpressure of RCS (110% of RCS design pressure, or 2,750 psia)
Operation	During Condition I and II events	During Condition III, IV, and beyond design basis events
Non-escalation requirement	Control grade PORVs cannot meet the n*on-escalation requirement.*	Not applicable. (1) The n*on-escalation requirement* is already violated by the time the PSVs

	It is possible to upgrade PORVs to safety grade, and thereby meet the *non-escalation requirement.*	open. (2) Cannot escalate beyond Condition IV.
Water relief capability	It is possible to qualify PORVs to relieve water.	PSVs, even if water-qualified, would not open until after the n*on-escalation requirement* is violated.
Single Failure	Control grade PORVs: GDC 21 does not apply. Safety grades PORVs: a stuck-open PORV can be closed or isolated.	GDC 21 cannot be met, since a stuck-open PSV cannot be isolated.

The PORVs, as well as pressurizer spray and heaters, comprise the pressurizer pressure control system. They are designed to prevent unnecessary reactor trips, and unnecessary challenges to the PSVs. The PORVs are designed to relieve enough pressure to keep the plant online during AOOs (e.g., turbine trips and partial load rejections). Westinghouse design accounts for this function. Some Westinghouse plants, known as *full load rejection plants*, are equipped with three PORVs. These plants, are capable of keeping the plant online following a turbine trip at full power.



The figure shows:

- One PORV is shown to open at 2,350 psia. This is enough to prevent the opening of any PSVs.
- The high pressure reactor trip (marked in red) will occur when pressure exceeds 2,400 psia.
- The PSV opening setpressure is higher than the pressure boundary that defines an AOO (i.e., the 2,400 psia reactor high pressure trip setpoint).
- If pressure continues to increase, after the reactor is tripped, and a PORV does not open, then one or more PSVs would open at 2,500 psia (set to the RCS design pressure).
- Pressure relief would be provided by the PORVs or by the PSVs; but not both.
- Three PSVs have more than six times the steam relief capacity of one PORV.
- AOOs would lie to the left of the red reactor trip line (i.e., pressures below 2,400 psia). If pressure exceeds the reactor trip setpoint (i.e., despite the reactor shutdown), then the reactor

shutdown has not accommodated the event. Consequently, the event is not an AOO. So events that lie to the right of the red line are not AOOs. If these events are not AOOs, then they must be Condition III or IV events.

- PSVs cannot open until after the AOO becomes a more serious event (i.e., opening of the PSVs cannot meet the *non-escalation requirement*).

In [1], Westinghouse states, *a water-solid pressurizer condition should be precluded when the pressurizer is at or above the set pressure of the PSRVs. An exception to this criterion can be made if the utility can support a position that their PSRVs are designed and qualified to relieve subcooled water.* When the pressurizer is at or above the set pressure of the PSVs, it means that the AOO has not been mitigated by a reactor shutdown. It has developed into a more serious event. Therefore, the *non-escalation requirement* is already violated before it is possible to open any PSVs. Qualification of the PSVs to relieve water, as Westinghouse suggests, has no bearing, whatsoever, upon their effectiveness in mitigating AOOs!

The NSAL [1] states that inoperative control systems, like the PORVs, are a conservative assumption. This is not always true. In analyses that are designed to demonstrate compliance with the *non-escalation requirement*, it is conservative to assume that the PORVs will operate. Compliance with the *non-escalation requirement* is demonstrated by showing that the PORVs would not relieve water, or else by qualifying them for water relief duty, and upgrading them to safety grade status. This was done in about half a dozen PWRs. The first of these were the Salem units, in 1997. [8] Upgrading the PORVs had been recommended, by the NRC, ten years earlier, *Based on a review of the failure events, it is concluded that the greatest safety benefit could be achieved by using PORV designs that are resistant to sticking open. Upgrading the PORVs, operators, or control components to safety-grade status could each effect a reduction in PORV failures.* [13]

Application of PSVs in lieu of PORVs, if it were possible, requires a repurposing of the PSVs. The PSVs are designed to operate during Condition IV accidents, like feedline breaks, and beyond design basis events, like anticipated transients without scram (ATWS) [10], where RCS overpressure is the sole issue. Once opened, the PRSVs will have fulfilled their RCS overpressure safety function. The PSVs are not designed to relieve water, and then reseat, or even to open during AOOs. The PRSVs cannot be used during AOOs, as water-qualified pressure relief devices, without first making significant design changes. This would be the responsibility of the PSV supplier, not the customer. In this case, Westinghouse's NSAL does not propose a plan, in detail, that could qualify the PSVs for such functions.

Qualifying the PSVs for water relief would be very difficult if the temperature of the water relieved is too low. This issue prevented Diablo Canyon, Units 1 and 2, from qualifying their PRSVs for water relief, in 1998 [11]. This question was also addressed, nine years later, in NSAL-07-10 [9], which stated, s*ubcooled water relief through the pressurizer safety valves (PRSVs) could potentially cause damage to the valves, rendering the RCS boundary unisolable.* PG&E opted, instead, to upgrade their PORVs to safety grade level, for their Diablo Canyon Units, in 2004 [12].

Furthermore, each PSV is about twice the size of a PORV. If the PSVs cycle open and closed, like PORVs, it is possible that one or more of them will be damaged. This can result in significant leakage, through the improperly reseated PSVs. The damaged PRSVs could leak by up to 200 gpm. This cannot be an acceptable outcome for an AOO. This might not even be considered to be a leak, since 200 gpm is about the flow capacity of a charging pump, at 2500 psia.

(b) A stuck-open PORV or PSV is a LOCA, not a leak

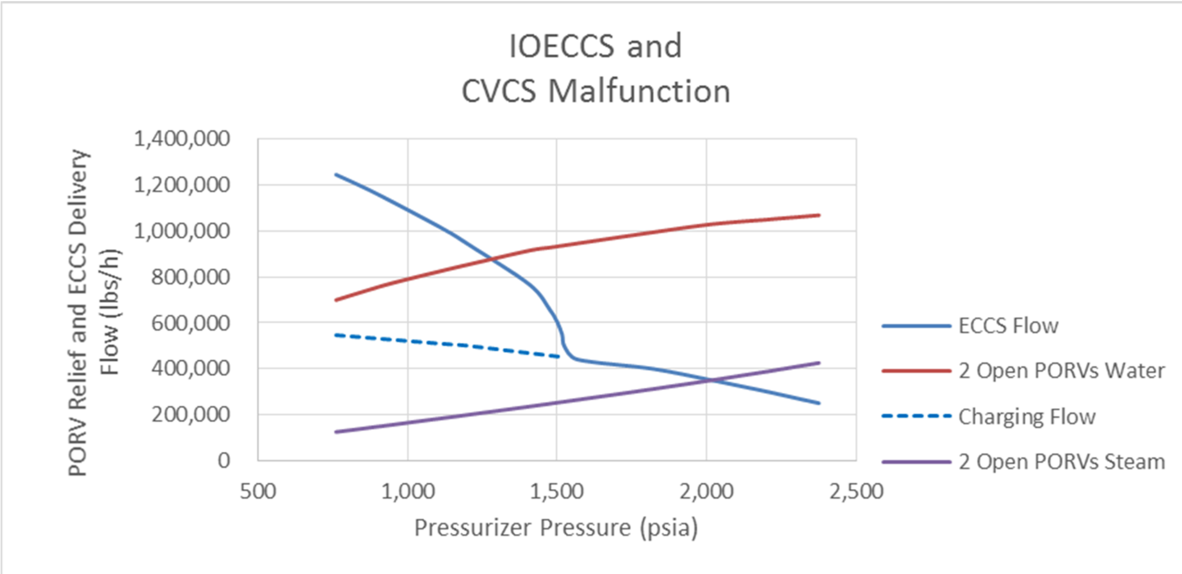
Westinghouse states, *American Nuclear Society standard 51.1/N18.2-1973 … describes … a condition II event as a "minor reactor coolant system leak which would not prevent orderly reactor shutdown and cooldown assuming makeup is provided by normal makeup systems only." … normal makeup systems are defined as those systems normally used to maintain reactor coolant inventory under respective conditions of startup, hot standby, power operation, or cooldown, using onsite power.* [1]

The ECCS is not a normal makeup system. The charging pumps, when actuated by an SI signal, cannot be considered to be a normal makeup system. This charging flow is not controlled by a pressurizer level program or influenced by letdown flow rates. It operates, simply, at maximum capacity, and it does not shut down until the operator shuts it down. That is, when the charging pumps are actuated by an SI signal, their function is to supply emergency core cooling, not to maintain a programmed pressurizer water level.

Westinghouse also states, *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).*

The situation that Westinghouse describes, wherein the pressurizer relief rate would be matched by the flow that is delivered by the ECCS flow, would be true only after RCS pressure has dropped to very low levels, late in an IOECCS or CVCS Malfunction scenario; long after the *non-escalation requirement* has been violated. At high RCS pressures, the water relief flow, will be critical (choked) water flow, which would be much greater than the ECCS flow that is delivered.

Here is an illustration of this relationship for two stuck-open PORVs during an IOECCS or a CVCS Malfunction. A CVCS Malfunction could be caused by an operator error, or failure of the controlling pressurizer water level sensor (in the low direction). Either of these could start both charging pumps, run them at maximum capacity, and ultimately fill the pressurizer. The following figure is a conservative depiction, since (1) the mass addition rate is high (i.e., the maximum ECCS and charging flow rates are represented), and (2) the PORV relief rate is the low, ASME-rated steam relief capacity for the two stuck open PORVs. (Some Westinghouse plants are equipped with three PORVs).



The figure shows that

(1) ECCS (i.e., charging) flow rate will not exceed the PORV steam relief rate until the RCS pressure drops to about 2,000 psia,
(2) ECCS flow rate will not exceed the PORV water relief rate until the RCS pressure drops to about 1,300 psia, and
(3) Charging flow rate will not exceed the PORV water relief rate until the RCS pressure drops below the accumulator injection setpoint (about 600 psia).
(4) One PSV has about twice the steam relief capacity of a PORV, and there are three of them. To estimate the steam relief rate of three stuck-open PSVs, triple the depicted steam relief rate of two stuck-open PORVs.

The pressurizer, during mass addition AOOs (e.g., IOECCS and CVCS Malfunction), is not an open system, like a bucket that is running over. It is a closed, pressurized system with a hole in it. The flow exiting through the hole is determined by the size of the hole, and the pressure of the system, not by the flow that may be entering the system. The NSAL's *mass-balance* argument (i.e., what goes in, comes out) contradicts the basic principles of fluid mechanics. Certain Westinghouse customers have adopted this rationale, and copied it into their license amendment requests (LARs), which they submitted to the NRC under Oath and Affirmation.

(c) Begging the question is not a valid licensing rationale

A plant that relies upon the PSVs to operate, in lieu of PORVs, must somehow keep the PORVs closed, as the IOECCS or CVCS Malfunction causes pressure to rise to the PPORV opening setpoint. PORVs that do not open cannot stick open. Therefore, assuming the PORVs remain closed, during an IOECCS, begs the question.

One customer, Exelon, convinced the NRC staff that water-qualified PSVs may be used during an IOECCS, in a license amendment request (LAR) for a power uprating. [7] Exelon shifted the pressure mitigation duty to the PSVs by claiming the PORVs would not operate (a non-conservative assumption). So, Exelon did not demonstrate that their plant designs' complied with the *non-escalation requirement,* by showing that their PORVs would not stick open. They just assumed their PORVs would not stick open.

(d) The false comparison

Westinghouse makes a false comparison when it states in [1]: *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.*

Westinghouse repeats this assertion, more than 14 years later, in [9]: *One acceptance criterion for an ANS Condition II event is that it should not generate a more severe plant condition. If the pressurizer becomes water solid, the potential exists for water to be discharged through the pressurizer PORVs or PSVs. If sustained subcooled water relief through the PSVs occurs, the valves may potentially be damaged such that the RCS pressure boundary may not remain intact. Should this occur, the consequences of these events could progress to resemble those associated with a more severe event, i.e., a small break loss-of-coolant accident (LOCA) in the pressurizer vapor space. While this is a violation of one of the acceptance criteria for a Condition II event, it is not considered a significant safety concern. Discharge of coolant out of the pressurizer relief system, especially for a post-reactor trip condition, is less severe than other ANS Condition ill LOCA events currently analyzed. This is particularly true considering that the pressurizer is located on the hot leg and this break location is less severe than the cold leg breaks considered in the licensing basis LOCA safety analyses. Due to the fact that the possible consequences are bounded by an ANS Condition III LOCA event, it is concluded that the potential non-conservatism of modeling the PORVs does not represent a SSH* (significant safety hazard) *or a failure to comply which would result in a SSH.*

These events cannot be compared because there is no common basis for comparison. Consider the differences:

	IOECCS with stuck-open PORV	Small (hot leg break) LOCA
Cause	Operator error or spurious signal	Rupture or mechanical fault
ANS event category	Condition III with a Condition II frequency of occurrence	Condition III
Analysis objectives: To show that the event will not …	Cause any fuel-clad damage, or Develop into a Condition III or IV event	Cause serious fuel-clad damage, or Develop into a Condition IV event
Initial RCS pressure	PORV opening (2350 psia)	Nominal pressure (2250 psia)
Reactor trip	Immediate; as part of the ECCS actuation signal	Demanded by a low pressurizer pressure condition
Duration of analysis	Until ECCS flow is manually ended (minutes)	Until RCS conditions are stabilized, with the core covered, and adequate decay heat removal is established (hours)
Relief or break flow	Steam (then water, if the pressurizer fills)	Water

ECCS flow	Maximum (all pumps operate)	Minimum (after the worst single failure is assumed)

Westinghouse's comparison is based solely upon predicted consequences. The two events also have different frequencies of occurrence, acceptance criteria, and blowdown characteristics. Furthermore, they're analyzed according to different objectives. One is based upon a conservatively high ECCS flow, to fill the pressurizer quickly, and the other is based upon a conservatively low ECCS flow, to impair core cooling effectiveness. In this aspect, Westinghouse loses sight of the analysis objectives by pronouncing the lower ECCS flow, used in the LOCA analysis, to be conservative for both events.

Furthermore, if the RCS were to pressurize during a Condition IV event, then PRSVs could open, by design, and relieve either steam or water. They could stick open, and create a 3.7 inch diameter hot leg LOCA. This would be an unisolable hole. PSVs sticking open during Condition IV events are of little or no concern, since (1) the *event escalation* requirement does not apply to Condition IV events (i.e., it's not possible to escalate an event beyond Condition IV), and (2) stuck open PRSVs are just one more problem, in a host of problems that are posed to the operators during a once-in-a-plant's-lifetime (or less) Condition IV accident.

(e) The SSH (significant safety hazard)

Consider, too, how a stuck-open PORV would be diagnosed. A stuck-open PORV would be indicated after the RCS has depressurized to below the PORV closing setpressure, and the PORV has not reseated. This would not occur until after the cause of the RCS pressurization, the ECCS flow, is ended. Therefore, there would be a stuck open PORV with no ECCS flow. The situation resembles the Three Mile Island accident of 1979. It would be necessary for the operators to isolate the open PORV, or somehow restore ECCS flow. The situation could actually exceed the Three Mile Island accident, since both PORVs would stick open after relieving water. One can argue that only one operator error separates this event from a TMI scenario.

The NSAL [1] states, *If ECCS flow is not terminated before water is discharged through the PSRVs, it cannot be demonstrated without plant specific PSRV operability assessments that this accident does not lead to a more serious plant condition. …. If ECCS flow is not terminated before the pressurizer becomes water solid and water is discharged through the PSRVs, it cannot be demonstrated that this accident does not lead to a more serious Condition III LOCA event.*

Recall that (1) PSVs do not open during AOOs, and (2) PSVs do not open if one of the PORVs is open. If the PORVs are conservatively assumed to operate, as designed, then the postulated operation of the PSVs would be moot.

(f) Operation of the PSVs is irrelevant

The NSAL [1] states, *it is assumed that PSRVs must not pass water in order to ensure their integrity and continued availability. Therefore, the Westinghouse internal event criterion for this Condition II event is revised such that subcooled water discharge through the Pressurizer Safety Relief Valves shall be precluded for a Condition II transient.*

Hence, a water-solid pressurizer condition should be precluded when the pressurizer is at or above the setpressure of the PSRVs. An exception to this criterion can be made if the utility can support a position that their PSRVs are designed and qualified to relieve subcooled water.

(g) Operation of the PORV bock valves could be a concern

The NSAL [1] also states, *Water relief through the PORVs is not a concern, because the PORV block valves can be used to isolate the PORVs if they fail to close.* The NSAL [1] does not advise customers must first to qualify their PORVs to relieve water, and upgrade their PORVs' automatic control system circuitry. It merely allows that PORVs might stick open. The PORV block valves can always be used to isolate any stuck open PORVs. Such an action must, necessarily, be taken after the *non-escalation requirement* has already been violated (i.e., a stuck-open PORV has transformed the IOECCS into a Condition III LOCA). In other words, isolating a PORV is an operation that is undertaken to mitigate a Condition III SBLOCA, not a Condition II IOECCS. [2]

(h) The PSVs cannot comply with GDC 21 when closure of the PSVs is required

If the PRSVs are repurposed, for use during AOOs, it becomes necessary to consider the possibility of a PSV failing to close, since the PRSVs would be required to close, as well as open. GDC 21 requires that, r*edundancy and independence designed into the protection system shall be sufficient to assure that … no single failure results in loss of the protection function*. One failed-open PSV would create a Condition III LOCA, which would violate the *non-escalation design requirement.*

GDC 21 is a design requirement that applies to protection systems that are used to deal with all events, from AOOs to Condition IV accidents (e.g., major LOCAs). The GDC 21 requirement is not linked to the specifics of single failures. For example, the ECCS is designed to flood the core after a major LOCA. One of two ECCS trains is assumed to fail, as per GDC 21, and the remaining train is required to fulfill the safety function of both ECCS trains. There is no specification as to why or how an ECCS train will fail. The ECCS train is assumed to fail because that would reduce the level of redundancy. One PSV will stick open for any reason. Therefore, the PSVs' reliability in relieving water (e.g., water qualification), no matter how high it may be, has nothing to do with meeting GDC 21.

In simple terms, PSVs that are connected in parallel, and cannot be isolated, cannot be assumed to close without at least one failing open. Such a system could not possibly meet the GDC 21 single failure requirement when all PSVs are required to close.

(i) Safety significance

Some of the advice, given in the NSALs, if implemented by the addressees, could create an SSH, as defined by 10 CFR §92. For example, the NSALs' suggestion that water-qualified PSVs may be applied during AOOs is not consistent with [5], and Westinghouse's plant designs.

The Petitioner maintains that Westinghouse has failed to comply with 10 CFR §21 reporting requirements. Furthermore, some of the NSALs' advice, if implemented, could be a safety defect, since it could generate an SSH.

See Attachment (C) for more information regarding safety significance.

Attachment C

Safety Significance

Several of Westinghouse's NSAL suggestions could lead to significant safety concerns. These are discussed, below, in terms of 10 CFR §50.92, and 10 CFR §50.21.

According to 10 CFR §50.92

10 CFR §50.92, *Issuance of amendment*. Section (a) states, *in determining whether an amendment to a license … will be issued to the applicant, the Commission will be guided by the considerations which govern the issuance of initial licenses … to the extent applicable and appropriate.*

It goes on to state, *(c) The Commission may make a final determination … that a proposed amendment to an operating license … 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.

The Petitioner would to these questions in the following manner:

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

If PSVs are operated in lieu of PORVs then it would be necessary to impose additional design requirements upon the PSVs, since they would be expected to perform during the IOECCS, the CVCS Malfunction, and any other AOOs that pressurize the RCS to the PORV opening setpoint. Now, the RCS will have to pressurize to the PSV opening setpoint (i.e., the RCS design pressure) during each of these AOOs. The PSVs will also be required to reseat, after having relieved water. Therefore, a new failure mode has been introduced: failure of a PSV to reseat. If this occurs, the result will be a small, hot leg LOCA. This LOCA will be more frequent than the currently analyzed LOCA. The probability of this Condition III LOCA will increase to equal the sum of the probabilities of all the AOOs that currently cause the RCS to pressurize to the PORV opening setpoint.

The consequences of the initiating AOOs are also increased, since operation of the PSVs, at 2500 psia, will always be required during most instances in which the PORVs would currently open. The consequences of stuck open PRSVs, if they occur, would be greater than the consequences of stuck open PORVs, since PSVs are not isolable.

Operation of the PSVs during AOOs is not in their design basis. Frequent pressurization of the RCS to its design pressure could also be outside the RCS design basis.

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

Recall that the PORVs are designed to prevent unnecessary challenges to the PSVs. Preventing the PORVs from opening, and relying upon the PSVs to open in lieu of the PORVs, creates a new accident. For the purpose of this Petition, it can be called an *unnecessary challenge to the PSVs (UCPSV)*. This creates a new accident. This event pressurizes the RCS to the PSV opening setpoint (i.e., the RCS design pressure). The frequency of occurrence for this event would be the sum of the frequencies of occurrence of the several AOOs in which PSVs are opened, in lieu of PORVs.

Note that, in the 1980s, the NRC was concerned with the safety implications of the high frequency (at that time) of unnecessary automatic reactor trips that were incurred at operating plants. Industry, including the Petitioner, worked to reduce the incidence of unnecessary automatic reactor trips.

The UCPSV would occur at pressures that are greater than the AOO defining boundary. Therefore, the UVPSV would be a Condition III event with the frequency of occurrence of an AOO. It could pose a greater threat to the public health and safety than the unnecessary automatic reactor trip, since the consequences of this event could be potentially greater. For example, a stuck open PSV would have a about twice the relief capacity of a PORV, and it would not be isolable. It could also exceed the number of allowable pressurizations of the RCS.

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

When the PORVs are applied, the margin of safety, to RCS overpressure is 400 psi (2750 psi minus 2350 psi). When the PSVs are applied, the margin of safety to RCS overpressure is reduced to 250 psi (2750 psi minus 2500 psi). The margin of safety is thus reduced by 37.5%.

If PSVs are required to perform, instead of PORVs, then some ANS Condition III events could occur with the combined frequencies of occurrence of certain AOOs. This increase in the total frequency of occurrence for ANS Condition III events (e.g., from once in the plant's lifetime to one or more times per year of operation) is also a reduction in safety margin.

According to 10 CFR §21

The NSAL [1] states, *the Inadvertent ECCS Actuation at Power event may progress from a Condition II to a more severe Condition III LOCA event …. 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.* Thus, Westinghouse dismisses *non-escalation requirement*, as *one of the applicable licensing basis criteria for a Condition II event,* and avoids addressing the safety significance that a failure to comply creates. The *non-escalation requirement* lies at the core of the event classification system upon which nuclear plant design and safety analysis are based.

The Condition III LOCA event that results from a stuck-open PORV or PSV is a Condition III event with the frequency of occurrence of an AOO. This would be a special Condition III LOCA event that is outside the plant's design basis. Since it could easily violate the acceptance criteria that are specified for Condition II events, it could be an event of high safety significance.

Consider the statements that Westinghouse, the Vendor, has included in its NSALs:

Westinghouse is unable to determine whether a defect causing a substantial safety hazard or a failure to

Comply resulting in a substantial safety hazard exists because sufficient plant specific information is not available. Under 10 CFR 21.21(b), if Westinghouse determines that there is insufficient information available to provide the capability to perform an evaluation, then Westinghouse must inform affected licensees of this determination. [1]

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(b). The NRC has not been notified of this issue. [1]

One acceptance criterion for an ANS Condition II event is that it should not generate a more severe plant condition. If the pressurizer becomes water solid, the potential exists for water to be discharged through the pressurizer PORVs or PSVs. If sustained subcooled water relief through the PSVs occurs, the valves may potentially be damaged such that the RCS pressure boundary may not remain intact. Should this occur, the consequences of these events could progress to resemble those associated with a more severe event, i.e., a small break loss-of-coolant accident (LOCA) in the pressurizer vapor space. While this is a violation of one of the acceptance criteria for a Condition II event, it is not considered a significant safety concern. [9]

Westinghouse has not reported this issue to the USNRC. [9]

The NSALs [1] [9] suggest that Westinghouse customers (i.e., licensees) may disregard their written licensing commitments to comply with design requirements, if non-compliance will not lead to significant safety concerns. As recently as 2007 [9], Westinghouse stated*, the potential non-conservatism of modeling the PORVs does not represent a SSH or a failure to comply which would result in a SSH.* Twenty years earlier, the NRC had stated, *a severe challenge to safety could occur in a PWR if the PORV sticks open-and the BV (block valve) fails to close, that is a small break LOCA.* [13]

Thus, Westinghouse claims that a failure to comply is acceptable if it would not lead to an SSH. Westinghouse also fails to advise its customers that instances of non-compliance must be approved by the NRC (i.e., through the Relief Request process), on a case-by-case basis.

Westinghouse claims an ignorance of the licensing basis analyses of some of its NSAL addressees to excuse itself from reporting this issue to the NRC, under the requirements of 10 CFR §21. Neither of the cited NSALs [1] [9] contain any information that is specific to the licensing basis analyses of any of the addressees. All of the subject plants, foreign and domestic, employ charging pumps, in their ECCS designs, that are capable of pressurizing their reactor coolant systems to their PORV and PSV opening pressure setpoints. That is the only plant-specific design or analysis feature that applies, and that feature is standardized among Westinghouse plant designs. PORV and PSV relief capacities, and setpressures are uniform throughout Westinghouse plant designs. The valves are sized according to plant power ratings.

Attachment D

Summary and Conclusion

Westinghouse is advising certain of its customers to operate its supplied equipment outside its design capabilities; in a manner that cannot meet applicable design requirements. This advice is rendered in at least two problematic publications (NSALs). [1] [9]. This Petition identifies problems in at least nine specific aspects of Westinghouse's advice.

In summary, here is a recount of the issues in this Petition, supplemented by some illustrative examples, questions, observations, and concluding remarks.

(a) PSVs may not be used in lieu of PORVs

In [1], Westinghouse states, *a water-solid pressurizer condition should be precluded when the pressurizer is at or above the set pressure of the PSRVs. An exception to this criterion can be made if the utility can support a position that their PSRVs are designed and qualified to relieve subcooled water.*

At least one customer, [16], Exelon states, *The SI flow refills the pressurizer until the pressurizer is water solid, and the SI flow results in liquid discharge through the pressurizer safety relief valves. The performance of the pressurizer safety valve system and the loads on pressurizer safety valves, associated piping, and supports as a result of liquid discharge through the pressurizer safety valves, was determined to be acceptable.*

Westinghouse fails to distinguish between the design and functions of control systems and safety systems. PORVs are control systems, and PSVs are safety systems. In an automobile, the PORVs would be seat belts, and the PSVs would be air bags. PORVs, like seat belts, are used often, to protect the driver during abrupt stops and occasional fender benders. PORVs and seat belts, can be engaged, disengaged, and even disconnected. The air bags, on the other hand, are always available, and used just once (maybe) in a car's lifetime, to protect the driver during a head-on collision. However, Westinghouse's air bags are special. After deployment, they can be stuffed back into the steering wheel, to be used again, and again, i.e., as often as necessary, to protect the driver, and passengers. Basically, Westinghouse suggests its customers need not bother buckling their seat belts, since they can always rely upon their air bags for protection.

(b) A stuck-open PORV or PSV is a LOCA, not a leak

Exelon has also implemented this suggestion, with success [16]. In its UFSAR analysis of the IOECCS, Exelon states, *American Nuclear Society standard 51.1/N18.2-1973 … describes … a condition II event as a "minor reactor coolant system leak which would not prevent orderly reactor shutdown and cooldown assuming makeup is provided by normal makeup systems only." …. normal makeup systems are defined as those systems normally used to maintain reactor coolant inventory under respective conditions of startup, hot standby, power operation, or cooldown, using onsite power. 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).*

(c) Begging the question is not a valid licensing rationale

Westinghouse suggests that its customers may demonstrate that their plant designs can meet the *non-escalation requirement* by begging the question. In this circular logic, the assumed outcome (i.e., that the PORVs will not stick open) becomes the actual outcome, since the PORVs are not permitted to open. PORVs that do not open cannot stick open. This gives new meaning to *AOO* (Assumed Outcome is the Outcome). This is the definition of circular reasoning. Oddly, this rationale has won the approval of the NRC in at least two applications. [7]

(d) The false comparison

Westinghouse attempts to apply the results of one accident analysis to envelope the results of another, less severe, accident analysis. Except the two analyses are not all comparable. Westinghouse claims that a hot leg LOCA will envelope an IOECCS or CVCS Malfunction.

There is a variation to this comparison. Instead of the hot leg LOCA, the comparison is made against the Inadvertent opening of Pressurizer Safety or Relief Valve. For example, Exelon's Byron and Braidwood UFSAR states, *The Inadvertent Operation of the ECCS during Power Operation event does not progress into a stuck open Pressurizer Safety Valve LOCA event. All three valves may lift in response to the event, but they will reclose. The resulting leakage from up to three pressurizer safety valves that are seated is bounded by flow through one fully open valve. The consequences of the event are bounded by the analysis described in UFSAR Section 15.6.1, "Inadvertent opening of Pressurizer Safety or Relief Valve".* [16] Exelon compares the (seated) PSV water leakage to the steam relief of the inadvertently opened valve in UFSAR Section 15.6.1. There are many other differences that would invalidate the comparison; but this one is sufficient.

(e) The SSH (significant safety hazard)

Westinghouse advises its customers, who hold NRC-issued operating licenses, that compliance with design requirements, upon which their licenses are based, can be set aside if the non-compliance will not cause an SSH.

Westinghouse advises its customers that, *the Inadvertent ECCS Actuation at Power event may progress from a Condition II to a more severe Condition III LOCA event …. 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.* [1]

There are only three *applicable licensing basis criteria for a Condition II event:* (1) no RCS overpressure, (2) no fuel clad damage, and (3) no progression to a more serious event (i.e., the *non-escalation requirement*). If failure to comply with either of the first two criteria would be *considered a significant safety concern,* then why wouldn't failure to comply with the third also be *considered a significant safety concern*? Compliance with the *non-escalation requirement* is as important as compliance with the other two criteria, since the *non-escalation requirement* establishes barriers between event categories that prevent events in low consequence, high frequency categories from developing into events in high consequence categories. Failure to design plants in accordance with *non-escalation requirement* could create new (unanalyzed) events of high consequence; and also with high frequency. This would undermine the event categorization scheme, upon which nuclear plant design, analysis, and regulation have been based, since the early 1970s.

(f) Operation of the PSVs is irrelevant

Operation of the PSVs, whether water-qualified or not, is irrelevant, since they cannot prevent an AOO from developing into a more serious event. PSVs will not open until after the AOO has developed into a more serious event.

Even if the opening of the PSVs were somehow made to become a feasible option, their operation would be unwise, since they would supply six times the relief capacity that is needed to mitigate an AOO. Furthermore, any relief or leakage from damaged PSVs would be unisolable.

(g) Operation of the PORV bock valves could be a concern

The NSAL [1] states, *Water relief through the PORVs is not a concern, because the PORV block valves can be used to isolate the PORVs if they fail to close.* The PORV block valves can be used to isolate any stuck open PORVs. However, the operators would not realize that a PORV is stuck open until after the RCS pressure has dropped below the PORV's closing setpoint, after the ECCS or CVCS flow has been ended. By this time, the *non-escalation requirement* has already been violated. The operators would be mitigating a Condition III LOCA, not a Condition II IOECCS. [2] Closing the block valve can be an effective response to a Condition III LOCA; but it cannot demonstrate compliance the *non-escalation requirement.*

If the block valves are closed before the ECCS or CVCS flow has been ended, then the RCS can pressurize to the PSV opening setpoint (and thereby transform the AOO into a more serious event), open the PSVs, and incur the problems that accompany operation of the PSVs. For example, if the PSVs are opened, after the ECCS system is shutdown, and a PSV fails open, then the situation would resemble a hot leg LOCA without ECCS, and raise TMI-like concerns.

(h) The PSVs cannot comply with GDC 21 when closure of the PSVs is required

If the customer adopts Westinghouse's suggestion, to qualify the PSVs for water relief, and apply them in lieu of PORVs, then it must qualify them to close, after relieving water, as well as open, and relieve steam. When the PSVs are repurposed, in this way, they must meet the GDC 21 single failure requirement for closing, as well as opening. GDC 21 requires that, r*edundancy and independence designed into the protection system shall be sufficient to assure that … no single failure results in loss of the protection function*. One failed-open PSV would constitute a loss of its protection function, since it would create a Condition III LOCA (i.e., a violation of the *non-escalation requirement*)*.*

Since the PSVs are connected in parallel, with no provisions for isolation, they cannot possibly meet the GDC 21 single failure requirement. Failure of even one PSV to close would result in a violation of the *non-escalation requirement.* Therefore, operation of the PSVs, in lieu of the PORVs, if it were feasible, would require the licensee to request a relief exception, from the NRC, regarding compliance with the single failure requirements of GDC 21 with respect to the PSVs' closure function.

(i) Safety significance

10 CFR §50.92 tests whether a proposed license amendment would:

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

(2) Create the possibility of a new or different kind of accident; or
(3) Involve a significant reduction in a margin of safety.

If a customer implements one or more of Westinghouse's suggestions, then the customer may not be able to truthfully assert that no significant hazards would result. This petition identifies certain of Westinghouse's recommendations as strategies that would create significant hazards (e.g., using PSVs instead of PORVs, disregarding compliance with the *non-escalation requirement* or failure to comply with GDC 21).

According to 10 CFR §21, Westinghouse is a major supplier of two basic components:

1. basic component means a structure, system, or component, or part thereof that affects its safety function necessary to assure: (A) The integrity of the reactor coolant pressure boundary

2. In all cases, basic component includes safety-related design, analysis, inspection, testing, fabrication, replacement of parts, or consulting services that are associated with the component hardware, design certification, design approval

The component, in this Petition, is the PSV, which protects the *integrity of the reactor coolant pressure boundary.*

Another component, in this Petition, is *safety-related design, and analysis,* that Westinghouse writes into a customer's FSAR or LAR, and the *consulting services* that Westinghouse supplies via its NSALs.

This Petition identifies certain, fundamental problems with the PSVs, supplied by Westinghouse, and its consulting services that suggest the PSVs be operated outside their design capabilities. Yet, Westinghouse creatively construes the requirements of 10 CFR §21 to transfer the reporting requirements to its customers. 10 CFR §21 states, *If the deviation or failure to comply is discovered by a supplier of basic components, or services associated with basic components, and the supplier determines that it does not have the capability to perform the evaluation to determine if a defect exists, then the supplier must inform the purchasers or affected licensees within five working days of this determination so that the purchasers or affected licensees may evaluate the deviation or failure to comply, pursuant to § 21.21(a).*

The supplier, Westinghouse, has equipped its customers' plants with PORVs that are not qualified to relieve water. In some plants, wherein the operators are incapable of terminating unwanted ECCS or charging flow before their plants' pressurizers can fill with water, the PORVs will open and relieve water. PORVs that are not qualified to relieve water cannot be relied upon to reseat completely after having relieved water. In these plants, the PORVs are basic components with a safety defect. Failure of a PORV to close means that the plant design cannot meet the *non-escalation requirement.* It is hard to understand how the NRC could grant an operating license for a plant that cannot meet all of its design requirements.

Westinghouse knows that all of its PORVs not qualified to relieve water; but doesn't know which of its customers' plant designs would require its PORVs to relieve water. Westinghouse claims that it *does not have the capability to perform the evaluation to determine if a defect exists,* if it does not possess the customers' current licensing basis safety analyses. Consequently, Westinghouse deems that informing its customers, through its NSALs, discharges its 10 CFR §21 reporting responsibility.

In this case, the defect is the Westinghouse-supplied PORVs. PORVs that are not qualified to relieve water impose a time-critical operating requirement in the event of an IOECCS or CVCS Malfunction (i.e., to end ECCS or charging flow before the pressurizer can fill). Since there are no Westinghouse-supplied PORVs that are qualified to relieve water, all of Westinghouse's customers must meet the aforementioned time-critical operating requirement. Therefore, Westinghouse does not need to perform an evaluation to determine if a defect exists. The defect is the PORV itself.

How could a PORV, a component in the pressure control system, be considered a safety defect? Failure of the PORVs to open is considered in accident analyses, since control systems cannot be relied upon like safety systems. The accident analyses verify that the PSVs are capable of protecting the *integrity of the reactor coolant pressure boundary.* Thus, the *RCS overpressure requirement* is satisfied. The accident analyses also consider scenarios in which a PORV is assumed to spuriously open. This is one of the events that could initiate an AOO. The licensing basis analyses show that all the AOO acceptance criteria are met.

Suppose a PORV fails to close? This event is not identified as a potential initiator of an AOO. A PORV cannot fail to close until after it opens. The initiating event is whatever causes the PORV to open (e.g., an IOECCS or CVCS Malfunction). If a PORV's failure to close is the direct result of the failure that initiates the event, then the two failures are treated as one, as specified by [3], *multiple failures resulting from a single occurrence are considered to be a single failure*. Therefore, a PORV's failure to close (i.e., a single failure) could lead directly to a Condition III event (e.g., a hot leg LOCA). A failed control system, or control system component that initiates a Condition III event is an unanalyzed event, and possibly a significant safety hazard.

At least half a dozen of Westinghouse's customers have corrected the defect by qualifying their PORVs for water relief, and upgrading them for use as safety grade components. This is what the supplier should have done. Instead, the supplier sends two NSALs, full of faulty, even harmful, advice to all its affected customers; without informing the NRC.

The two NSALs, *per se,* may be a safety defect, as defined in 10 CFR §21, since any nuclear plant operator could obtain copies of these and other NSALs, perhaps from the NRC website[2], and implement one or more of the problematic suggestions they contain. It is not known how many other faulty NSALs exist.

Furthermore, the NRC staff has approved some LARs that are based upon the faulty Westinghouse NSAL recommendations. [7] The Petitioner requests the NRC staff to take steps to prevent the recurrence of such approvals.

The Petitioner requests the NRC staff to take the following actions:

(1) Publish a generic communication, e.g., a Generic Letter, to inform licensees of the problematic advice in Westinghouse's NSALs. [1] [9] Consider GL 79-45 as an example, and precedent. [14]

(2) Revise the applicable standard review plans (in NUREG-0800) to alert NRC reviewers to licensees' implementations of certain, unacceptable items of advice from Westinghouse's NSALs.

[2] These NSAL were obtained by the NRC staff when they were cited in several LARs that were under review.

(3) Require Westinghouse to submit all of its NSALs to the NRC staff for information, and review.
(4) Require Westinghouse to make the applicable NSAL retractions, and revisions, and file the relevant 10 CFR §21 reports.
(5) Require Westinghouse to copy the NRC staff in its distribution of future NSALs.
(6) Fine Westinghouse, an amount that is sufficient to reimburse the NRC staff for the taxpayers' money it spent to review LARs that were based upon faulty advice in Westinghouse NSALs.
(7) Fine Westinghouse, an additional amount, as a penalty for this longstanding, continuing deception.

The two NSALs, published over a span of 14 years, which encompasses a pertinent NRC Regulatory Issue Summary [2], could mislead addressees into adopting risky, even harmful interpretations of Westinghouse's advice, involving fantastic licensing strategies, unrealistic plant modifications, and irrational operating procedures. Furthermore, certain parts of these NSALs could lead licensees to make false statements to the NRC in their *no significant hazards statements*.

These false statements could also be in violation of the federal false statement statute (18 U.S.C. §1001(a)). Generally, courts require proof of five elements to sustain a conviction under the Act:

First- The defendant made a statement;
Second- The statement was false;
Third- The statement was material;
Fourth- The statement was within the jurisdiction of a government department or agency; and
Fifth- The statement was made knowingly and willfully.

Proving the first four elements would not be difficult. Proving the fifth element could be facilitated by the NRC's implementation of actions (1) and (2). If false statements are subsequently submitted to the NRC, then they could be reasonably construed to have been made knowingly, and willingly, and thereby become subject to the federal false statement statute (18 U.S.C. §1001(a)). [17]

References

[1] NSAL-93-013, G.G. Ament and K.J. Vavrek, Westinghouse ESBU, June 30, 1993, and NSAL-93-013, Supplement 1, J.S. Galembush, Westinghouse ESBU, October 28, 1994 (ADAMS Accession No. ML052930330)

[2] NRC RIS 2005-29, *Anticipated Transients that Could Develop into More Serious Events*, December 14, 2005 (ADAMS No. ML051890212)

[3] 10 CFR §50, Appendix A, 36 FR 12733, July 7, 1971

[4] ANSI/ANS-51.1-1983, *Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants*, April 29, 1983

[5] American Nuclear Society, *Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants*, ANS-N18.2-1973, La Grange Park, Illinois, August 6, 1973

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

[7] U.S. NRC, letter from George F. Dick, Jr., to Oliver D. Kingsley, Exelon Generation Company, LLC, “Issuance of Amendments; Increase in Reactor Power, Byron Station, Units 1 and 2, and Braidwood Station, Units 1 and 2 (TAC Nos. MA9428, MA9429, MA9426, and MA9427),” dated May 4, 2001, (ADAMS Accession No. ML033040016)

[8] U.S. NRC, Salem Nuclear Generating Station, Unit Nos. 1 and 2 (TAC Nos. M97827 and M97828) Amendment Nos.194 and 177, dated June 4, 1997 (ADAMS Accession No. ML011720397)

[9] NSAL-07-10, Loss-of-Normal Feedwater/Loss-of-Offsite AC Power Analysis PORV Modeling Assumptions, J.T. Crane and A.J. Macdonald, Westinghouse, November 7, 2007 (ADAMS No. ML100140163)

[10] Westinghouse Electric Corporation to U.S. NRC, ATWS SUBMITTAL, NS-TMA-2182, dated December 30, 1979 (ADAMS Accession No. ML041130109)

[11] LER 98-001-01, Diablo Canyon Units 1 and 2, Pacific Gas & Electric, Reactor Coolant System Outside Design Basis for Inadvertent Emergency Core Cooling System Actuation at Power Due to Non-Conservative Assumptions for Pressurizer Safety Valve Operation, October 22, 1998, (ADAMS No. 9810270409)

[12] Diablo Canyon Power Plant, Unit Nos. 1 and 2 - Issuance of Amendment Re: Credit for Automatic Actuation of Pressurizer Power Operated Relief Valves, USNRC (ADAMS No. ML041950260)

[13] NUREG/CR-4692, *Operating Experience Review of Failures of Power Operated Relief Valves and Block Valves in Nuclear Power Plants*, Oak Ridge National Laboratory, G. A. Murphy and J. W. Cletcher II, December, 1987 (ADAMS No. ML040230365)

[14] Transmittal of Reports Regarding Foreign Reactor Operating Experiences, September 25, 1979 USNRC, (ADAMS No. ML031320181 and GL 79-45)

[15] Commonwealth Edison Company to U.S. NRC, Request for a License Amendment to Permit Uprated Power Operations at Byron and Braidwood Stations, dated July 5, 2000, (ADAMS Accession No. ML003730544 and ML003730536)

[16] Exelon Generation Company, LLC, “Byron/Braidwood Nuclear Stations Updated Final Safety Analysis Report (UFSAR),” Revision 15, dated December 2014 (ADAMS Accession No. ML14363A495)

[17] See Morrison, supra note 23, at 113–14; United States v. Jiang, 476 F.3d 1026, 1029 (9th Cir. 2007); United States v. Robison, 505 F.3d 1208, 1226 (11th Cir. 2007); United States v. Pickett, 209 F. Supp. 2d 84, 87 (D.C. Cir. 2002) (citing United States v. Crop Growers Corp., 954 F. Supp. 335, 349 (D.C. Cir. 1997)); United States v. Kosth, 257 F.3d 712, 718 (7th Cir. 2001).