Westinghouse

Nuclear Safety Advisory Letter

This is a notification of a recently identified potential safety issue pertaining to basic components supplied by Westinghouse. This information is being provided so that you can conduct a review of this issue to determine if any action is required.

1000 Westinghouse Drive, Cranberry Township, PA 16066

Subject: **Impact of a Break in the Reactor Coolant Pump No. 1 Seal Leak-off Line Piping on Seal Leakage During a Loss of Seal Cooling Event**	Number: **NSAL-15-2**
Basic Component: No. 1 Reactor Coolant Pump Seal and Seal Leak-off Piping	Date: 03/23/2015
Substantial Safety Hazard or Failure to Comply Pursuant to 10 CFR 21.21(a) Transfer of Information Pursuant to 10 CFR 21.21(b) Advisory Information Pursuant to 10 CFR 21.21(d)(2)	Yes ☐ No ☐ N/A ☐ Yes ☒ Yes ☐

SUMMARY

During plant events that result in a loss of reactor coolant pump (RCP) seal cooling; such as a station blackout, extended loss of AC power, or fires that affect power supplies, the leakage through the RCP No. 1 seal increases. Seal leakage reduces the reactor coolant inventory and should be minimized, especially during these events. This behavior was initially investigated for a generic plant configuration, as documented in WCAP-10541, Revision 2 (Reference 1). Nuclear Safety Advisory Letter (NSAL) NSAL-14-1, Revision 1 (Reference 2) further evaluated the impact of various No. 1 leak-off line configurations on the seal leakage flow rate. The evaluation of seal leakage assumed that the leak-off line piping remained intact for the duration of the loss of seal cooling event. This is a key assumption because seal leakage for most leak-off line configurations was determined by analysis to be limited by the two-phase choked flow that occurs at the flow measurement device, whereas a break in the piping upstream of this choked-flow device could produce an even higher flow rate.

The plants that are affected are those with a structural analysis of the No. 1 leak-off line piping upstream of the flow measurement device that does not bound a pressure of 2045 psia at a temperature corresponding to the maximum cold leg temperature (as determined by the steam generator design pressure) possible during RCS natural circulation conditions. Licensees that have demonstrated, or can demonstrate, that the leak-off line piping can withstand these conditions without a break are not affected.

Additional information, if required, may be obtained from Leah Gussenbauer (412) 374-5316

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Risk Applications and Methods

As demonstrated in Table 1, plants which have ¾ inch piping that will remain in the flow path following a pipe break have flow rates which are bounded by those contained in Reference 2. These plants would not experience a substantial safety hazard even if the piping analysis determines that a piping break may occur.

Figure 1 shows a typical configuration of the No. 1 seal leak-off line. The piping of concern is upstream of the flow element and is indicated in red.

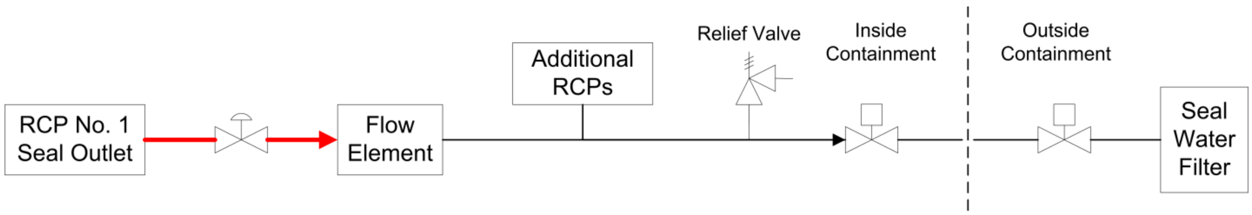

Figure 1: Typical Configuration of a No. 1 Seal Leak-off Line

The structural integrity of the No. 1 seal leak-off piping is typically the responsibility of the licensee. Technical Bulletin (TB) NSD-TB-91-07 (Reference 3) informed licensees that the pressure and temperature in the leak-off line may exceed the original design values as a result of various potential overpressure scenarios, including a loss of seal cooling. This TB recommended that licensees review their No. 1 seal leak-off line structural integrity analysis and modify it if necessary. However, due to the plant-specific nature of the scenarios, NSD-TB-91-07 did not specifically define the maximum pressure and temperature conditions that may occur in the piping following these events.

This NSAL addresses the Westinghouse RCP shaft seal package for a nominal 8 inch pump shaft size where the No. 1 seal is a controlled-leakage film-riding design and the No. 2 and No. 3 seals are rubbing-face designs. Plants with Westinghouse-style RCP seals other than those described above (such as those with the nominal 7 inch pump shaft size) may or may not be affected. This NSAL provides conservative No. 1 seal leakage flow rates if a break were to occur in the leak-off line piping upstream of the flow measurement device due to a short-duration pressure transient similar to that observed in the hot shock test at the Montereau test facility in France.

Westinghouse evaluated this condition and performed a flow rate analysis. Results indicated that if a pipe break upstream of the flow element were to occur due to pressurization from the increased seal leakage, but ¾ inch diameter piping were to remain in the seal leakage flow path, this event would not represent a substantial safety hazard and is not reportable in accordance with 10 CFR 21.

Westinghouse is transferring information contained in this NSAL pursuant to 10 CFR 21.21(b) for plants with 2 inch diameter seal leak-off piping. This is because the aforementioned conclusion for the ¾ inch piping may not be the same for the 2 inch piping configuration and because Westinghouse does not have design responsibility for the analysis of the piping and supports and, therefore, cannot evaluate the integrity of the leak-off system piping for each of those plants. Affected licensees should determine if a break in the 2 inch piping upstream of the No. 1 seal leak-off line flow element could occur for a pressure of 2045 psia and the maximum RCS cold leg temperature. The cold leg temperature is plant specific and depends on the steam generator design pressure. Table 3 provides a list of plants that may be affected by this issue.

ISSUE DESCRIPTION

The hot shock test performed in the 1980s at Montereau has been recently reviewed in detail due to ongoing seal leakage activities. This test is documented in Section 7, Appendix B and Appendix C of WCAP-10541 (Reference 1). The test used a Model 93D pump, a 7 inch aluminum oxide No. 1 seal, and a representative leak-off line. The results of this test indicated that a short duration high pressure transient occurred in the RCP No. 1 leak-off line following the loss of seal cooling conditions that were associated with the test. The pressure in the No. 1 leak-off line peaked at a value of 2045 psia and remained elevated for a few minutes. The transient pressure increase coincided with the abrupt transition to hot fluid (up to 536°F in the test) passing through the No. 1 seal while sub-cooled fluid remained in the leak-off line. The transient had a short duration because the leak-off line quickly transitioned to the steady state two-phase flow conditions discussed in References 1 and 2.

The presence of a short duration high pressure transient may not be accounted for in the current plant-specific piping structural analyses for the No. 1 leak-off line. The 2045 psia value reached during this loss of seal cooling event may exceed the value used in the current piping stress analysis, and pipe support analysis. Given these findings, the continued structural integrity of the leak-off line piping, which was a key assumption in References 1 and 2, may be compromised.

Note that the transient pressure value of 2045 psia is considered to be a conservatively high value. Affected licensees have the option to determine and justify a lower value for this short duration high pressure transient, or use the conservative value of 2045 psia.

TECHNICAL EVALUATION

Westinghouse gathered information on seal leak-off line configuration data from licensees in the United States that presently use Westinghouse nominal 8 inch shaft No. 1 RCP seals. Westinghouse reviewed the information with respect to the pipe diameters that are present upstream of the flow measurement device. The pipe diameter was chosen as the key parameter for this evaluation, because if a break were to occur upstream of the flow measurement device, the pipe diameter would be the most important parameter controlling the seal leakage flow rate. Piping components such as valves and pipe bends may reduce the flow rate, but the inner flow diameter of the piping is the controlling factor.

It was determined that nominal leak-off pipe diameters of ¾ inch and 2 inch at the exit of the pump are prevalent throughout the fleet. A small number of plants having 1 inch and 1½ inch piping upstream of the flow measurement device were also noted; however, licensees with these piping diameters also had either ¾ inch or 2 inch piping at the exit of the pump.

Westinghouse performed an analysis to determine conservative values for the flow rate that would result from a pipe break that may occur in either ¾ inch or 2 inch piping. The analysis assumed Schedule 160 piping, as a review of the leak-off line configurations indicated this is the pipe schedule that was used at the exit of the pump. The analysis was performed for a range of reactor coolant system (RCS) temperatures and pressures to represent the cool down and depressurization of the RCS following a loss of seal cooling. The analysis addressed No. 1 seals with silicon nitride faceplates, which result in lower leakage than the aluminum oxide faceplate material. The silicon nitride faceplate was chosen because it is expected that most, if not all, plants have upgraded to this material. Table 1 shows the results of this analysis. Note that the leak rate values in Table 1 are on a per RCP basis. While the values in Table 1 may differ for seals with aluminum oxide faceplates, the overall 10 CFR 21 conclusions would not change.

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Table 1: Seal Leakage Flow Rate For a Broken Seal Leak-off Line Pipe			
Nominal Leak-off Pipe Diameter (in.)	RCS Cold Leg Pressure (psia)	RCS Cold Leg Temperature (ºF)	Seal Leakage Flow Rate (gpm/pump)
¾	2250	572	24.2
	1500	572	27.4
	310	415	14.8
2	2250	572	71.2
	1500	572	330
	310	415	113

For the 2 inch configuration cases with 1500 psia and 310 psia inlet pressures, the Westinghouse analysis code used to calculate seal leakage does not converge to a solution. This may be due to limitations of the code, or the code could be accurately reflecting the limits of stable seal operation. Based on a review of seal stability using a simplified seal model, the 2 inch configuration may be unstable for the 1500 psia and 310 psia inlet conditions. Therefore, the values shown in Table 1 are calculated using only the resistance of the leak-off line. Although seal instability is expected to be a low probability event due to seal mechanisms not modeled by the Westinghouse analysis code, conservative flow rates are reported assuming the instability occurs because the analytical seal model does not converge.

SAFETY SIGNIFICANCE

The licensee will need to determine the structural impact of the transient peak pressure on its RCP No. 1 leak-off lines, because, as discussed previously, Westinghouse does not hold the analysis of record (AOR) for these lines. Additionally, for plants with the 2 inch leak-off piping configuration, the affected licensees will need to determine the impact on nuclear safety significance. While the Westinghouse analysis shows that a pipe break in a ¾ inch line will not exceed the worst-case leak rates provided in NSAL-14-1, Revision 1, Westinghouse cannot conclude the same for the larger 2 inch diameter piping. Additionally, licensees may have specific design criteria that would mitigate this issue, such as plant modifications performed in response to Reference 3 or use of an alternate seal cooling system for loss of normal seal cooling events.

If the No. 1 leak-off line piping integrity is maintained during the short duration pressure transient, the seal leakage analysis documented in NSAL-14-1 remains applicable. A substantial safety hazard does not exist, i.e., is not reportable in accordance with 10 CFR 21, for licensees that can demonstrate that there will be no break in the leak-off line piping upstream of the flow measurement device.

The piping downstream of the flow measurement device is not critical for the case in which the flow measurement device is an orifice, since the two-phase flow chokes at the orifice. For other flow measurement devices which are not expected to have choked flow, the downstream piping does have an impact on the seal leakage flow rate. However, it was concluded that the plant configurations with flow measurement devices that are not orifices, contain ¾ inch piping upstream of the device such that the Table 1 conclusions for the ¾ inch piping configuration are applicable to a downstream pipe break.

The seal leakage flow rate values in Table 1 for the ¾ inch leak-off line piping configuration are lower than the limiting case in Table 2a of NSAL-14-1. Therefore, the safety significance determination in that NSAL remains valid. A substantial safety hazard does not exist, i.e., is not reportable in accordance with 10 CFR 21, for licensees that can demonstrate that if a break occurs in the leak-off line piping, it will occur in a location such that ¾ inch piping remains in the flow path upstream of the pipe break.

The seal leakage flow rate values in Table 1 for the 2 inch leak-off line piping configuration are higher than the limiting case in Table 2a of NSAL-14-1. Therefore, an evaluation has been performed to assist licensees in determining the safety significance of a break for this configuration. The following discussion is only applicable to licensees with 2 inch leak-off piping if they should experience a break such that ¾ inch diameter piping does not remain in the seal leakage flow path.

RCS Loop Operability in Modes 1, 2, 3, and 4

RCS loop operability in Modes 1, 2, 3, and 4, which is required to provide forced reactor coolant circulation, is not impacted. This issue only affects the amount of flow through the RCP seal package following a loss of all seal cooling. The RCP seal package performance during normal operation of the RCPs is not impacted. In loss of all seal cooling events, reactor trip and shutdown of all RCPs either occurs via operator actions or automatically. Therefore, no at-power implications could exist as a result of this issue.

10 CFR 50.63 Station Blackout (SBO)

The Westinghouse nuclear steam supply system (NSSS) designs are licensed with an SBO coping time based on NUMARC 87-00, Revision 1 (Reference 4). These coping times typically fall into the range of 4 to 8 hours for recovery of AC power. NSAL-14-1 (Reference 2) demonstrates that the time to core uncovery for the limiting configuration if the leak-off line piping for all RCS loops remains intact is greater than the 8 hour coping time. For the seal leakage values following a break in 2 inch piping shown in Table 1, the time to core uncovery, based on a reference plant analysis, is estimated to be approximately 1½ to 2 hours because the emergency core cooling system (ECCS) is not available. This estimate is based on a Westinghouse standard four loop plant design. Because the RCS volume of Westinghouse-designed NSSSs follow a quasi-linear relationship, this time estimate is considered to be generally applicable to three and two loop designs as well. Note that this estimate assumes that the short duration pressure spike results in a break in the leak-off line piping for all RCS loops. If a break were to occur in fewer associated loop leak-off lines or the flow area of the failed pipe was smaller due to the pipe diameter, the time to core uncovery would be increased. Therefore, it is concluded that core uncovery following an SBO event with breaks in all No. 1 leak-off lines would occur prior to a plant licensing basis coping time of 4 hours.

Extended Loss of AC Power (ELAP)

Plant-specific FLEX (diverse and flexible coping strategies) analyses evaluate timelines necessary to implement operator actions, including RCS makeup capabilities to maintain core cooling. In these ELAP analyses, the time to initiation of reflux cooling and to core uncovery will be reduced if a break were to occur in one or more leak-off lines. This may result in additional requirements for makeup capabilities or resources to complete the required manual actions to maintain core cooling. If this issue is determined to be applicable, each plant will need to review their ELAP analyses.

10 CFR 50.48 Fire Protection Program

The requirements of an acceptable fire protection program are specified in 10 CFR 50.48. The potential impacts on fire protection for plants that have implemented a risk-informed fire program in accordance with National Fire Protection Association (NFPA) Standard 805 are discussed in the next section covering Risk-Informed Applications and Methods. For plants not implementing NFPA 805, 10 CFR 50 Appendix R is generally applicable. Section (III)(L)(2)(b) of 10 CFR 50 Appendix R states "The reactor coolant makeup function shall be capable of maintaining the reactor coolant level above the top of the core for BWRs [boiling water reactors] and be within the level indication in the pressurizer for PWRs [pressurized water reactors]." Some fire scenarios at some plants can result in a loss of seal cooling and a loss of all high pressure RCS makeup capability. The Appendix R fire protection coping analyses evaluate timelines necessary to restore some means of high pressure makeup to maintain the pressurizer water level on span. The seal leakage values in Table 1 may challenge those timelines and may not allow the pressurizer water level to remain on span for both ¾ inch and 2 inch configurations.

Risk-Informed Applications and Methods

Probabilistic risk assessment (PRA) models have been used as part of the justification for making changes to nuclear power plant design and operation in accordance with Regulatory Guides 1.174 through 1.178. Examples include risk-informed changes to the Technical Specifications Completion Times and Surveillance Frequencies, as well as changes to the plant fire protection licensing basis under 10 CFR 50.48(c) through NFPA 805. The PRA models used in risk-informed applications typically include an assessment of the risk of core damage due to loss of RCP seal cooling events. In some cases, the loss of RCP seal cooling can be a significant contributor to the overall risk of core damage calculated by the PRA model.

WCAP-15603 (Reference 5) provides the current PRA methodology for modeling seal leakage from a Westinghouse-designed RCP seal package following loss of all seal cooling. This approach, which has been reviewed and approved by the U.S. Nuclear Regulatory Commission (NRC), is commonly referred to as the WOG2000 model. As summarized in Table 2, the WOG2000 model includes probabilities for a range of potential leak rates considering functionality of the No. 1 and No. 2 seals.

Table 2: WOG2000 PRA Model for RCP Seal Behavior Following Loss of All Seal Cooling

Probability[1]	Leak Rate (gpm/pump)	No. 1 Seal Status	No. 2 Seal Status
0.79	21	Normal[2]	Normal
0.01	76	Failed	Normal
0.1975	182	Normal	Failed
0.0025	480	Failed	Failed

1. Probability given a loss of seal cooling event.
2. Normal refers to expected behavior for a loss of seal cooling event.

A comparison of the data in Tables 1 and 2 shows that the assumed 21 gpm leakage flow rate with functional No. 1 and No. 2 seals in the WOG2000 model significantly underestimates the predicted seal leakage flow rate in plants with 2 inch leak-off line piping in the event of a pipe break. Since this scenario is the most probable in the WOG2000 model (79% probability), a significant increase in the associated leakage flow rate could have a significant effect on the core damage frequency (CDF) and dominant risk contributors predicted by the PRA model.

If a licensee determines that a break in the leak-off line piping and the associated flow rates in Table 1 could occur following a loss of RCP seal cooling, the licensee should review any risk-informed applications based on the PRA model to determine whether this issue affects the conclusions of those risk-informed applications implemented by the licensee.

Reportability

The issue discussed in this NSAL does not affect licensees that can demonstrate that a break will not occur in the No. 1 leak-off piping upstream of the flow measurement device. For these plants, this issue is not reportable in accordance with 10 CFR 21.

This issue could adversely impact licensing basis evaluations (but would not be 10 CFR 21 reportable) for licensees that determine that if a break occurs in the leak-off line piping, it will occur in a location such that ¾ inch piping remains in the flow path upstream of the pipe break. The seal leakage flow rate values in Table 1 for the ¾ inch leak-off line piping configuration are lower than the limiting case in Table 2a of NSAL-14-1 (Reference 2). Therefore, the safety significance determination in NSAL-14-1 remains valid. For these licensees, the issue would not constitute an unanalyzed condition that degrades plant safety as discussed in Section 3.2.4 of NUREG-1022, Revision 3 (Reference 6). This issue is also not reportable in accordance with 10 CFR 21 for these licensees. Note that for plants that have flow measurement devices other than orifices, this conclusion is also applicable for a break in downstream piping if the ¾ inch piping integrity remains in the flow path upstream of the break.

This issue could adversely impact licensing basis evaluations for licensees that conclude that a break may occur in their 2 inch No. 1 seal leak-off piping not containing ¾ inch piping in the flow path upstream of the pipe break. For these licensees, the issue has the potential to represent an unanalyzed condition that degrades plant safety as discussed in Section 3.2.4 of NUREG-1022 (Reference 6), unless mitigation features are available. The Table 1 flow rates following a break in the 2 inch leak-off line piping could result in a time-to-core uncovery of approximately 1½ to 2 hours for station blackout, which is less than the licensing basis coping time of 4 hours (or 8 hours). The potential for core uncovery to occur in this timeframe may require evaluation to determine whether a substantial safety hazard exists. However, Westinghouse does not have the design responsibility or hold the AOR for the seal leak-off line piping, and cannot determine if a break would occur following a loss of seal cooling event. Westinghouse is transferring this information to the affected licensees pursuant to 10 CFR 21.21(b).

AFFECTED PLANTS

The affected plants are those that have a Westinghouse-supplied shaft seal package design where the No. 1 seal is a controlled leakage film riding seal and the No. 2 and No. 3 seals are both rubbing-face designs. Plants with Combustion Engineering (CE) -NSSS are not affected. Successful actuation and sealing of the Westinghouse **SHIELD**® passive thermal shutdown seal mitigates the short duration pressure transient for the RCP No. 1 seal leak-off line with which the shutdown seal is associated. Successful functioning of an alternate seal cooling system can also mitigate the issue by preventing a loss of seal cooling from occurring. However, it is recommended that licensees that have installed the **SHIELD** shutdown seal or alternate seal cooling systems review this issue and evaluate the impact of the pressure transient on the leak-off line. Doing this evaluation will conservatively ensure seal leak-off line integrity will be maintained in the unlikely event of an equipment malfunction that causes these means of mitigation to be ineffective.

Specifically, the plants that are affected are those with an analysis of the No. 1 leak-off line piping upstream of the flow measurement device that does not bound a pressure of 2045 psia at a temperature corresponding to the maximum cold leg temperature (as determined by the steam generators) possible during RCS natural circulation conditions. Licensees that have demonstrated, or can demonstrate, that the leak-off line piping can withstand these conditions without a break are not affected. As demonstrated in Table 1, plants which have ¾ inch piping that will remain in the flow path following a pipe break have flow rates which are bounded by those contained in Reference 2. These plants would not experience a substantial safety hazard even if the piping analysis determines that a piping break may occur.

Plants which have 2 inch piping that will remain in the flow path (without ¾ inch piping also in the flow path) following a pipe break may have break flow rates that are not bounded by NSAL-14-1 (Reference 2). These plants are affected by this issue.

The plants listed in Table 3 originally were supplied with an RCP shaft seal package by Westinghouse. Table 3 includes plants with both ¾ inch and 2 inch seal leak-off piping. In some cases, the current RCP seals may have been supplied by a manufacturer other than Westinghouse. The plants listed in Table 3 should review this NSAL and determine if they are affected by the issue.

Table 3: RCP Seals Originally Supplied by Westinghouse

A.W. Vogtle 1 & 2	Diablo Canyon 1 & 2	McGuire 1 & 2	Sequoyah 1 & 2
Almaraz 1 & 2	Doel 1, 2 & 4	Mihama 1, 2 & 3	Shearon Harris
Angra 1	Genkai 1	Millstone 3	Sizewell B
Asco 1 & 2	H.B. Robinson 2	North Anna 1 & 2	South Texas 1 & 2
Beaver Valley 1 & 2	Hanbit 1 & 2	Oconee 1	Surry 1 & 2
Beznau 1 & 2	Ikata 1	Ohi 1 & 2	Takahama 1 & 2
Braidwood 1 & 2	Indian Point 2 & 3	Point Beach 1 & 2	Three Mile Island 1
Byron 1 & 2	J.M. Farley 1 & 2	Prairie Island 1 & 2	Turkey Point 3 & 4
Callaway	Kori 1, 2, 3 & 4	R.E. Ginna	V.C. Summer
Catawba 1 & 2	Krsko	Ringhals 2, 3 & 4	Vandellos 2
Comanche Peak 1 & 2	Lemoniz 1 & 2	Salem 1 & 2	Watts Bar 1 & 2
D.C. Cook 1 & 2	Maanshan 1 & 2	Seabrook 1	Wolf Creek

NRC AWARENESS

Westinghouse has not notified the NRC of this issue. However, the NRC is aware of the potential for a loss of seal cooling transient pressure peak in the No. 1 seal leak-off line piping due to its interaction with some plant licensees and with the Pressurized Water Reactor Owners Group (PWROG) on issues associated with an ELAP.

RECOMMENDED ACTIONS

1. Evaluate the piping in the No. 1 seal leak-off line between the pump and the flow measurement device to determine if this piping remains intact for a pressure of 2045 psia, coincident with the plant-specific maximum RCS cold leg temperature.

   a. The AOR, using the code of construction methodology for faulted stresses, should be updated as necessary to demonstrate continued code compliance with the increased pressure. InfoGram IG-15-1 (Reference 7) provided initial guidance to assist licensees in determining if the piping would remain intact.

   b. Should it be determined that a pipe break or burst could occur in 2 inch No. 1 seal leak-off piping upstream of the flow measurement device, and if effective means of mitigation does not exist, the possibility exists for a substantial safety hazard to occur. For plants in this category, Westinghouse is transferring this information pursuant to 10 CFR 21.21(b) because Westinghouse does not have design responsibility for the leak-off line piping stress analysis and supports.

2. If the aforementioned analysis shows that the piping may break, for either ¾ inch or 2 inch pipe, and effective means of mitigation do not exist, evaluate the impact to the current licensing basis and other in-process programs:

   a. 10 CFR 50.63 station blackout: review the impact of the issue to determine if the updated time to core uncovery remains greater than the plant specific coping time.

   b. FLEX mitigating strategies: review the impact of the issue against the timelines and capabilities in the plant specific analyses.

c. 10 CFR 50.48 (a) and (b) and 10 CFR 50 Appendix R fire protection: review the impact of the issue against the fire protection program licensing basis.

d. Risk-informed applications: review risk-informed applications to determine whether this issue affects the conclusions of any risk-informed applications in use by the plant.

3. Wherever any portion of the leak-off line integrity upstream of the flow measurement device cannot be demonstrated to satisfy applicable design basis structural analysis limits, plant licensees should consider making a physical change to ensure piping integrity.

4. Plant specific emergency and abnormal operating procedures should be reviewed for consistency with the recommended actions in TB-15-1 (Reference 8).

REFERENCES

1. Westinghouse Report WCAP-10541, Revision 2, "Westinghouse Owners Group Report, Reactor Coolant Pump Seal Performance Following a Loss of All AC Power," November 1986.
2. Westinghouse Nuclear Safety Advisory Letter NSAL-14-1, Revision 1, "Impact of Reactor Coolant Pump No. 1 Seal Leakoff Piping on Reactor Coolant Pump Seal Leakage During a Loss of All Seal Cooling," September 8, 2014.
3. Westinghouse Technical Bulletin NSD-TB-91-07, Revision 1, "Overpressurization of RCP #1 Seal Leakoff Line," June 18, 1992.
4. NUMARC 87-00, Revision 1, "Guidelines and Technical Bases for NUMARC Initiatives Addressing Station Blackout at Light Water Reactors," August 1991.
5. Westinghouse Report WCAP-15603, Revision 1-A, "WOG 2000 Reactor Coolant Pump Seal Leakage Model for Westinghouse PWRs," June 2003.
6. NUREG-1022, Revision 3, "Event Report Guidelines 10 CFR 50.72 and 50.73," January 2013.
7. Westinghouse InfoGram IG-15-1, Revision 0, "Elevated Transient Pressure in Reactor Coolant Pump No. 1 Leak-off Line Following a Loss of All Seal Cooling," February 27, 2015.
8. Westinghouse Technical Bulletin TB-15-1, Revision 0, "RCS Temperature and Pressure Limits for No. 2 RCP Seal," March 3, 2015.

Approved for Release by WEC (see ML15105A102)