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REVIEW OF NUCLEAR POWER REACTOR COOLANT SYSTEM LEAKAGE EVENTS AND LEAK DETECTION REQUIREMENTS

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ABSTRACT

In response to the vessel head event at the Davis–Besse reactor, the U.S. Nuclear Regulatory Commission (NRC) formed a Lessons Learned Task Force (LLTF). Four action plans were formulated to respond to the recommendations of the LLTF. The action plans involved efforts on barrier integrity, stress corrosion cracking (SCC), operating experience, and inspection and program management. One part of the action plan on barrier integrity was an assessment to identify potential safety benefits from changes in requirements pertaining to leakage in the reactor coolant system (RCS). In this effort, experiments and models were reviewed to identify correlations between crack size, crack–tip–opening displacement (CTOD), and leak rate in the RCS. Sensitivity studies using the Seepage Quantification of Upsets In Reactor Tubes (SQUIRT) code were carried out to correlate crack parameters, such as crack size, with leak rate for various types of crack configurations in RCS components. A database that identifies the leakage source, leakage rate, and resulting actions from RCS leaks discovered in U.S. light water reactors was developed. Humidity monitoring systems for detecting leakage and acoustic emission crack monitoring systems for the detection of crack initiation and growth before a leak occurs were also considered. New approaches to the detection of a leak in the reactor head region by monitoring boric–acid aerosols were also considered.

Keywords: Leaks, leak detection, leakage models, reactor coolant boundary.

1. INTRODUCTION

One of the primary challenges to the integrity of the reactor coolant pressure boundary (RCPB) is stress corrosion cracking (SCC). Primary water stress corrosion cracking (PWSCC) of Alloy 600 nozzles was first observed in pressurizer instrument nozzles in a number of U.S. reactors in 1986. In 1991, similar cracking was found in vessel head penetration (VHP) nozzles at a French plant (Bugey). At the time, it was thought that circumferential cracking of Alloy 600 nozzles was unlikely. However, in 2001, inspections at the Oconee Nuclear Power Station revealed significant circumferential cracks in VHP nozzles. As a result of the Oconee event and some subsequent findings at other reactors, the NRC issued Bulletin 2001–01, requesting information to verify that licensees were in compliance with existing regulations with respect to the integrity of the RCPB.

In March 2002, while inspections were underway in response to Bulletin 2001–01, three control rod drive mechanism (CRDM) nozzles with indication of through–wall axial cracking that resulted in RCPB leakage were identified at the Davis-Besse Nuclear Power Station. During the nozzle repair activities, the licensee removed boric acid deposits from the reactor vessel head (RVH), conducted a visual examination of the area, and identified a cavity (178 mm by 102-127 mm) at the widest part that extended down to the stainless steel cladding on the downhill side of nozzle 3. The extent of the damage indicated that it had occurred over an extended period, and that the licensee's programs to inspect the reactor pressure vessel (RPV) head and to identify and correct boric acid leakage were ineffective.

In response to the Davis-Besse event, the NRC formed a Lessons Learned Task Force (LLTF). The LLTF conducted an independent evaluation of the NRC's regulatory processes related to ensuring RVH integrity in order to identify and recommend areas of improvement applicable to the NRC and the industry. Four action plans were formulated to respond to the recommendations. The action plans involve barrier integrity, stress corrosion cracking, operating experience, and inspection and program management. One part of the action plan on barrier integrity is an assessment of potential safety benefits of changes in requirements pertaining to leakage in the reactor coolant system (RCS). The technical basis for the leakage requirements has been reviewed in Kupperman et al., (2004).

To ensure that the RCPB has not been breached because of a through–wall crack we currently rely on periodic inspection of RCPB components. An additional approach would be to monitor continuously the RCPB (or critical locations or components) by methods capable of detecting material degradation before leakage occurs. The two approaches could be implemented in combination to provide greater assurance of barrier integrity.

From a practical standpoint, the RCPB cannot be made completely leak–tight because some leakage will occur through pump and valve seals, etc. However, as part of a defense–in–depth philosophy for ensuring the integrity of the RCPB, improved leakage requirements (e.g., establishment of action requirements based on increases in unidentified leak rates, and more accurate identification, measurement, and collection of leakage from known sources to minimize interference with the detection of leakage from unknown sources) could better identify RCPB breaches. This knowledge would allow reactor operators to take action to prevent additional degradation of the pressure boundary. Such improvements could be achieved through additional requirements on the use of existing leak detection systems (e.g., reductions in the global leakage limits). However, existing systems may not be adequate to provide assurance that leakage is low enough to avoid boric–acid induced corrosion of carbon and low alloy steel components.

2. REVIEW OF RCS LEAKAGE EXPERIMENTS AND LEAK–RATE MODELS

The most frequently used software packages for predicting leak rates in reactor cooling system (RCS) piping components in the US are the PIpe Crack Evaluation Program (PICEP) (Norris et al., 1984) and SQUIRT (Paul et al., 1994, Scott et al., 2003) codes. Both use the same Henry–Fauske model for flow–though tubes to describe two–phase flow through a crack. Both codes can be used to compute two–phase flow rates through cracks in LWR piping systems given the material properties of the piping, thermohydraulic conditions under load, crack geometry, crack type, and orientation.

Three major sets of leak–rate experiments are available for comparison with the codes. The initial sets of data for the validation of the SQUIRT code included data on flow through capillary tubes, tight slits, and pipes with intergranular stress corrosion cracking (IGSCC). More recent sets of data were developed in Japan by Matsumoto et al. (1989) and in Canada by Boag et al. (1990). All of these tests involved two–phase flow of sub-cooled water through cracks or tight slits to simulate cracks with different roughness and crack openings.

Figure 1 shows results from the original validation of the SQUIRT code. The most pertinent data are from the IGSCC cracked–pipe tests, the other data are for slits or capillaries. Above 0.02 kg/s (0.32 gpm) most of the data are within a factor of 2 of the predicted values. Below 0.02 kg/s (0.32 gpm) the data can differ from the predicted values by factors of +10 and –5.

The Japanese experiments (Matsumoto et al., 1989) were conducted primarily with plate test specimens with well-defined crack opening displacement (COD), crack length, and surface roughness values. Tests were also done with water or steam in pipes with fatigue cracks. These data are compared with the predictions of the SQUIRT and PICEP () codes (Norris et al., 1984) in Fig. 2. These are the only data available for the flow of saturated steam. SQUIRT and PICEP appeared to give reasonable predictions for leak rates of about 0.02-0.05 kg/s (0.3 to 0.8 gpm). At leak rates above ≈ 0.13 kg/s (2 gpm), the analyses under predict the experimental leak rates. The PICEP is marginally more conservative than SQUIRT at the lowest leak rates. Under–prediction of the leak rate is conservative for leak before break (LBB) analysis or leak detection.

The Canadian experiments (Boag et al.. 1990) employed circumferential crack geometries that were precise, straight–sided, and smooth–surfaced. Some single–phase leak rate tests were performed. The experimental results are compared with the SQUIRT predictions in Fig. 3. Most of the predicted values are within a factor of ± 20% of the observed values, with more experimental data scatter below a leak rate of 0.15 kg/s (2.3 gpm).

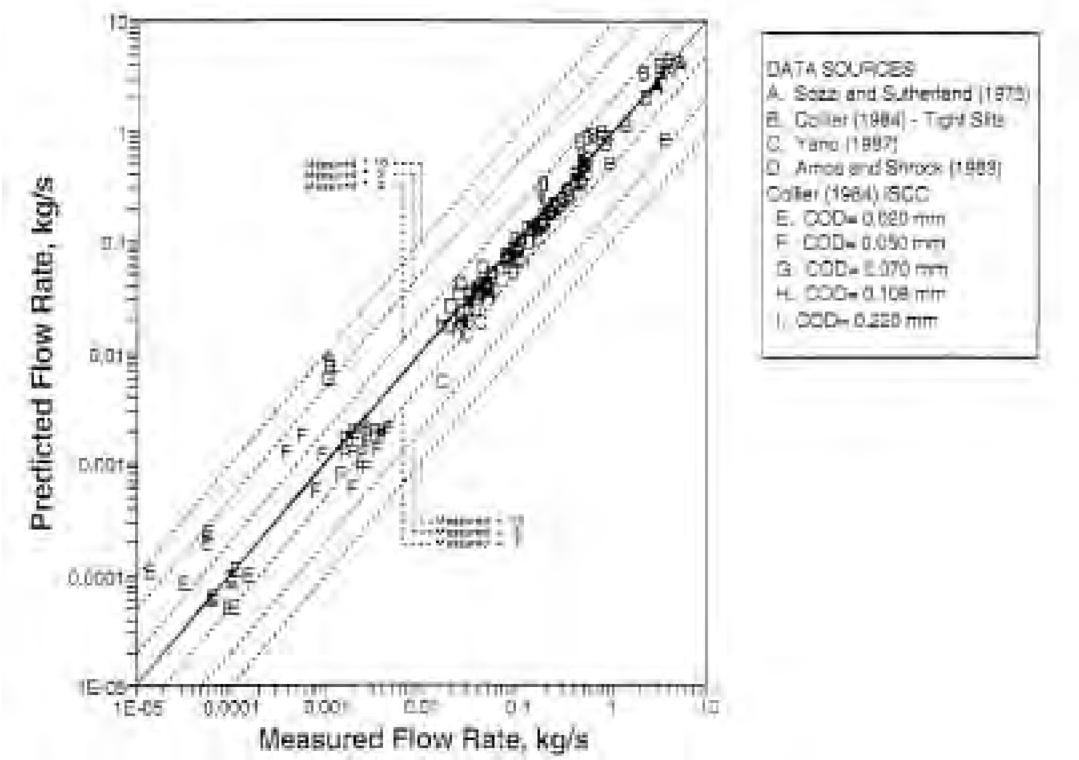

Figure 1. Comparison of two–phase flow leakage rate tests used to validate the initial SQUIRT model.

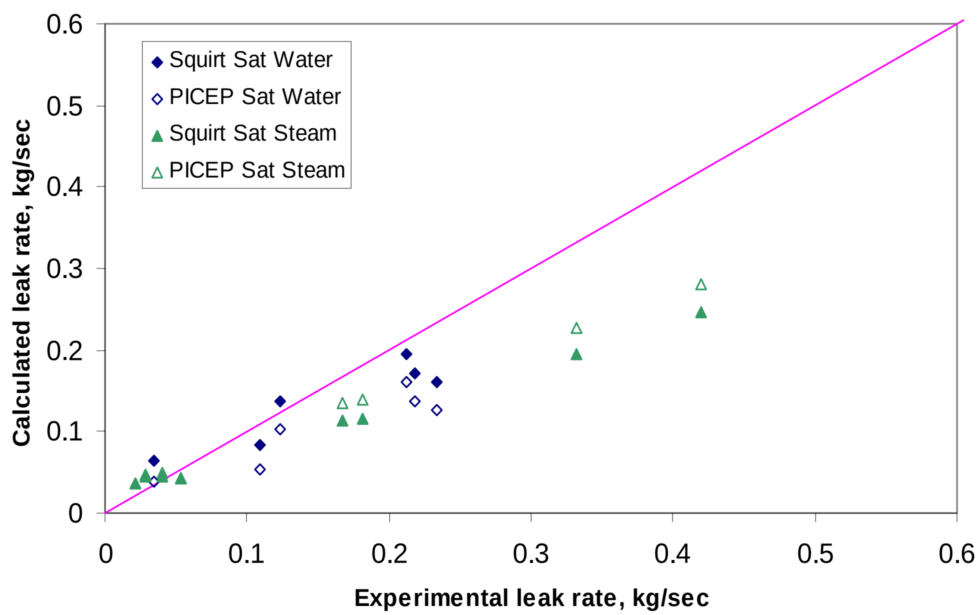

Figure 2. Comparison *of PICEP- and SQUIRT-predicted leakage rates with experimentally measured leakage rates from Japanese leakage–rate experiments Matsumoto et al. (1989).*

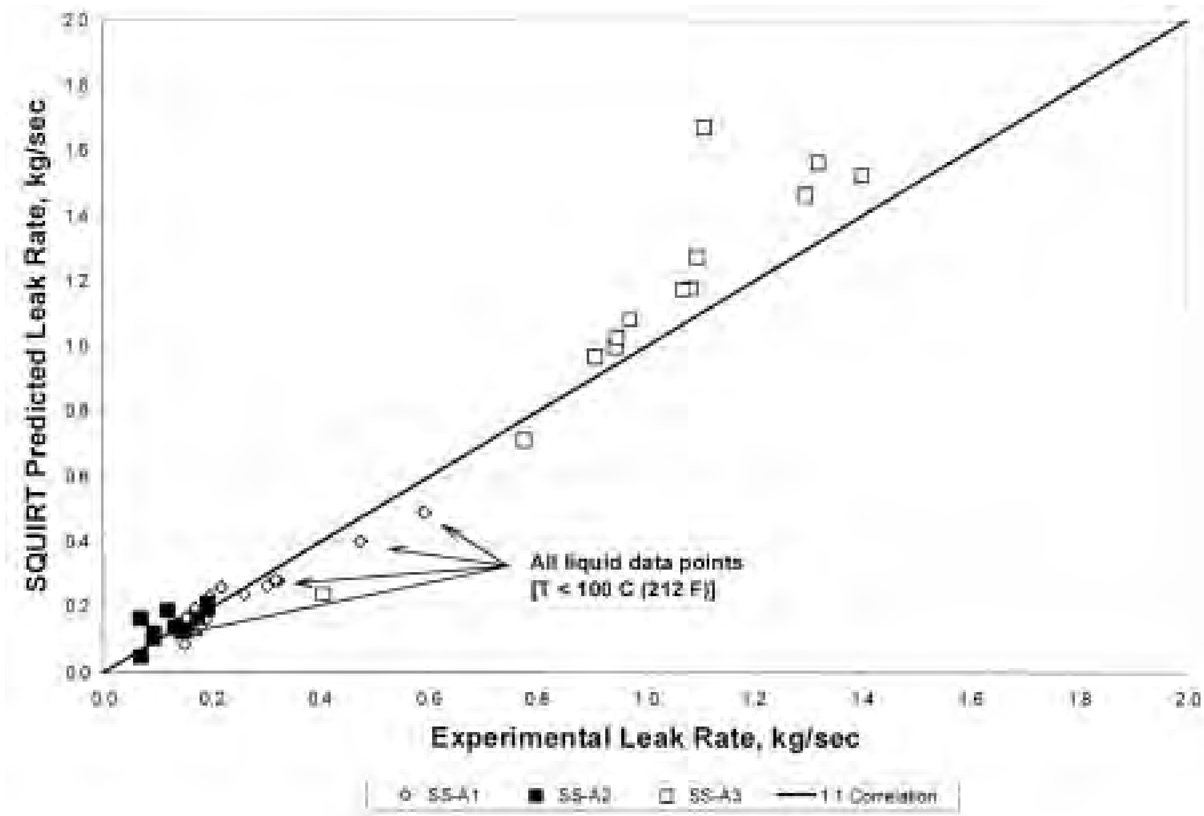

Figure 3. Comparison of SQUIRT-predicted leakage rates versus experimentally measured leakage rates for Ontario–Hydro experiments Boag et al. (1990).

Recent NRC studies have shown that the selection of the crack morphology parameters, such as surface roughness, the number of turns the crack takes, and the ratio of the actual flow path length to the thickness of the pipe, describing the crack flow path has a significant impact on predicted leak rates. Based on a study of cracks removed from service, it was determined that these parameters are dependent on the crack–opening displacement (Rahman et al., 1995). For a very tight crack, i.e., very narrow crack opening, the appropriate roughness is that along the grain boundary, and as the crack goes from one grain to the next there could be a turn. In such a case the crack surface roughness is low, but the number of turns is high and the actual length of the flow path much greater than the thickness of the pipe. If the crack opening is very large compared to the grain size, the appropriate roughness corresponds to about half of the grain size, and there would be very few turns and the length of the flow path is close to the thickness of the pipe. Figure 3 shows that predictions using the COD–dependent relationships are in agreement with experimental data even for leak rates <0.013 kg/s (0.2 gpm).

Figure 4 illustrates the variability in leak rate due to the variability the crack morphology parameters. The ratio between the mean value and the 2–percent upper fractile is typically a factor of ~2. For the relatively large leak rate shown in Fig. 4, the variability in leak rates due to the variability in the crack morphology parameters is consistent with the scatter in the observed and predicted results shown in Fig. 1. However, the variability in the leakage rate due to variation in the morphology parameters is not sufficient to account for the scatter in the data at lower leak rates. For small cracks additional factors such as the effect of the difference between actual crack shape and the rectangular crack assumed in the models may have a significant effect.

3. BASIS FOR RCS LEAKAGE MONITORING REQUIREMENTS

NRC Regulatory Guide (RG) 1.45 issued in 1973, established capabilities for leak detection systems acceptable to the staff. The technical basis for the determination of the capabilities considered acceptable for leak detection systems is not clear, although the first draft of RG 1.45 stated that cracks leaking at a rate of 0.063 kg/s (1 gpm) would be smaller than critical size by a factor of at least two, based on limited analytical studies and experimental data. RG 1.45 proposes that leaks should be monitored to a sensitivity of 0.063 kg/s (1 gpm) or better with at least three

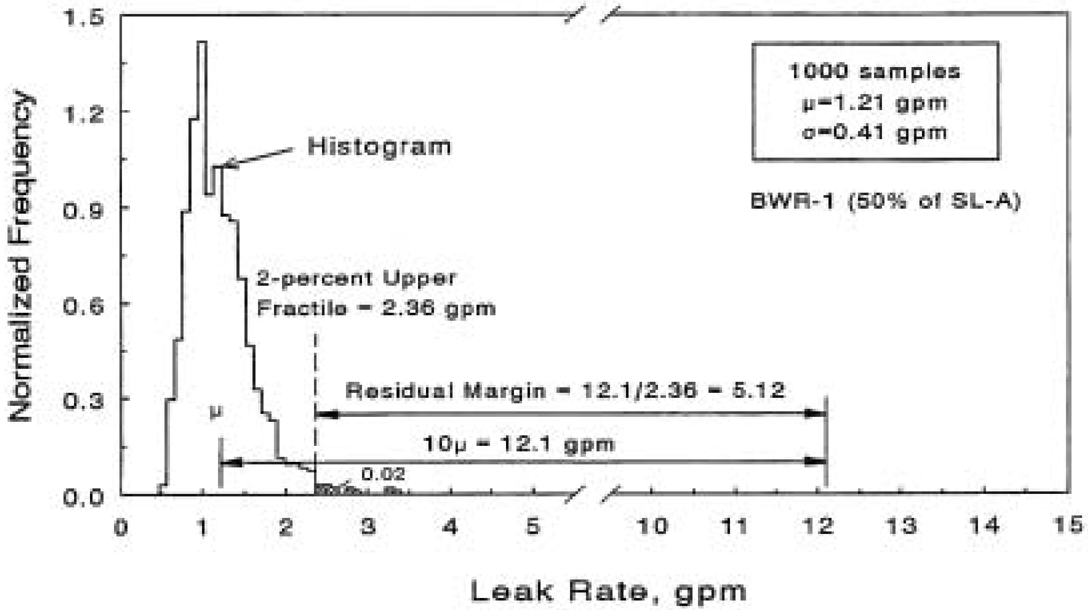

Figure 4. Histogram of leakage rate at 50% Service Level A for case BWR–1 in Rahman et al. (1995) [IGSCC crack case].

detection methods. The leak detection system should be able to detect a 0.063 kg/s (1 gpm) leak in less than 1 hour, and alarms for the leak detection systems should be located in the control room. Sump level and airborne particulate radioactivity monitors are required. The third method could be either a condensate flow monitor or any type of radiation. An airborne gaseous radioactivity monitor has been the usual choice. In RG 1.45 monitoring of the bulk humidity, temperature, and pressure in the containment are considered as indirect indications of leakage, but are not considered as one of the three detection methods.

In RG 1.45, for 1% and 0% failed fuel, it is assumed that a particulate monitor could detect a 0.063 kg/s (1 gpm) leak within 1 minute for both failed fuel assumptions, and that the gaseous monitor could detect a 0.063 kg/s (1 gpm) leak in about 2 minutes for 1% failed fuel and 100 minutes for 0% failed fuel. These conclusions were based on activity levels that are very high by current fuel performance standards, and hence, actual sensitivities and response times of such systems are worse than the estimates in RG 1.45. Sump pumps were estimated to be capable of detecting 0.063 kg/s (1 gpm) leaks within an hour.

Seven years after NRC Regulatory Guide 1.45 was published, the Instrument Society of America (ISA) issued the standard, ISA–S67.03–1982, "Standard for Light Water Reactor Coolant Pressure Boundary Leak Detection," (1982). It is a detailed, comprehensive document that can be a supplement or replacement for RG 1.45. The ISA position for the detection of leakage changes is also based on known capabilities at the time, and specifies that a 0.063 kg/s (1 gpm) increase in a PWR leak rate, and a 0.13-kg/s (2 gpm) increase in a BWR leak rate be detected within 1 hour. This standard also provides general equations for measurement sensitivities and response times of sump level and leakage flow monitoring. Equations for sensitivity and response times of radiation, humidity, and temperature monitors are also presented.

For a plant, the technical specifications (TSs) establish the requirements for leak monitoring systems and protocols for action when an anomaly is recognized. With respect to leakage, TSs are generally the same from plant to plant with some differences in the details. One of the first TS limits was established for the Monticello BWR in 1969. A limit of 1.6 kg/s (25 gpm) on identified leakage rate and a limit of 0.32 kg/s (5 gpm) on unidentified leakage rate was established. No documentation has been found on the technical basis used to establish these limits in 1969. The total allowed limit on leakage rate (identified plus unidentified) appears to have been based on the inventory makeup capability and sump capacity rather than RCS integrity assurance. Typical limits used today for PWRs are 0.063 kg/s (1 gpm) for unidentified leakage rate and 10 gpm (0.63-kg/s) total identified leakage rate. For BWRs, they are 0.32-kg/s (5 gpm) for unidentified and 1.6-kg/s (25 gpm) total identified leakage rates, with a capability to detect a leakage rate increase of 0.13-kg/s (2 gpm) within 24 h. Subsequent studies of the failure behavior of reactor coolant systems showed that for many piping systems these limits provide significant margin against gross failure of reactor piping to sustained stress loads. Margins increase with increasing pipe diameter and increasing loads under normal operation. The margins are larger for cracks due to corrosion fatigue than for cracks due to SCC. Detection systems such as the sump typically measure total leakage. To identify a leak with such a system, one must often compare the amount of leakage from all known sources to the total measured leakage. The difference between these two quantities is then the unidentified leakage.

The current PWR standard technical specifications require that for any leak in the RCS pressure boundary that cannot be isolated, if unidentified RCS leakage rate exceeds 0.063 kg/s (1 gpm) or if identified leakage rate exceeds 0.63 kg/s (10 gpm), the plant must be placed in hot standby (mode 3) within 6 hours and cold shutdown within the following 30 h. The evaluation related to safety should begin within four hours of detecting the leak. Two leak detection systems based on different principles, one capable of detection radiation, must be functioning when the reactor is operating. However, a radiation monitor can be inoperative for two days if two other leak detection systems are operating.

4. LEAKAGE OPERATING EXPERIENCE

We developed a U.S. plant leakage operating experience database that contains information since 1970 on LWR leakage events and leak detection systems. Sources for input to the database included Licensee Event Reports (LERs), NRC Information Notices through 2004, and NRC reports covering prior work, such as Shah et al. (1998), Kupperman et al. (1988), Kupperman et al. (1988b), and Hutton et al. (1991). Literature searches were carried out to identify other relevant publications (e.g., articles from *Nuclear Safety*) and databases such as an EPRI report co–sponsored by the Swedish Nuclear Power Inspectorate (SKI) on reactor piping failures (Bush et al., 1998). The Internet search engines Google and Yahoo were used to locate about 15 leak events not found in other sources. The database currently contains over 500 events dating from 1970.

The data for each leak include (a) the LER number if an LER is the source of information, (b) the location of leakage, (c) the leakage rate [actual leak rate if known, however, for many cases the actual leak rates are small (<0.0006 kg/s (0.01 gpm)) and not known precisely, although some qualitative information ("slowly dripping", etc.) may be available], (d) the operation of reactor when leakage was detected, (e) how the leakage was detected, (f) the basis for the decision that a leakage has occurred, (g) the time required to recognize there was an unidentified leakage, (h) the action that was taken, (i) the relevant nondestructive and destructive evaluation reports, (j) the cause of leakage, (k) the leakage requirements, (l) the crack type and size if crack was cause of leakage, and (m) any environmental impact. The data fields for leak detection systems included the (a) method of detection, (b) vendor for system, (c) sensitivity, (d) reliability, (e) response time, (f) accuracy, (g) estimated false alarm rate, (h) area of coverage, (i) maintenance required, (j) training required for its implementation, (k) calibration procedures, (l) site validation procedure, (m) experience under field conditions, and (n) source of information.

For leakage events in the current database, PWRs account for 70% of RCS leaks (about 66% of nuclear power plants are PWRs). In the 1988 report, "Assessment of Leak Detection Systems for LWRs," (Kupperman, 1988), PWRs accounted for 73% of the leak events. The relative rate of occurrence of leak events for PWRs and BWRs has not changed much since 1988. However, the total number of reported leaks identified in the current database has declined steadily from 48 in 1985 to 14 in 2003, as shown in Figure 5.

Figure 6 shows the relative frequency of leakage events associated with various components. "Seal" implies leaks from pump seals. "Pipe" leakage includes piping, lines, and small tubing but not leakages from steam generator tubes. "Valve" implies leakages from valves (packing or stem) that do not involve a leaking weld or valve body. "Weld" includes "pin–hole" leakages as well as cracks. "Nozzle" covers nozzles not associated with CRDM. "Sleeve" includes pressurizer heater sleeves. Leakages from welds represent nearly 20% of the leaks in the database. Valve leakages not involving a weld were another frequent source of leaks (18%). Leakages from piping account for 26% of the leaks. Leakages from all types of cracks are involved in over 40% of all leakage events in the database. In 19% of the PWR leakage events boric acid was visually observed at the site of the leakage.

Stress corrosion cracks were the source of the leakage more often than fatigue cracks in PWRs, while fatigue cracks were reported as the source of leaks more often in BWRs. When fatigue cracks were the cause of the leakage and the description noted more than just "fatigue" as the cause, "high–cycle fatigue" was mentioned much more often than "low–cycle fatigue," with "thermal–fatigue" noted only occasionally. For cracks with leakage rates >0.0063 kg/s (0.1 gpm), (excluding steam generators) fatigue cracks were noted as the type of crack about 50% more of the time than were SCC. This finding is consistent with the observation that SCCs are tighter and thus result in lower leakage rates compared to fatigue cracks for a given crack length.

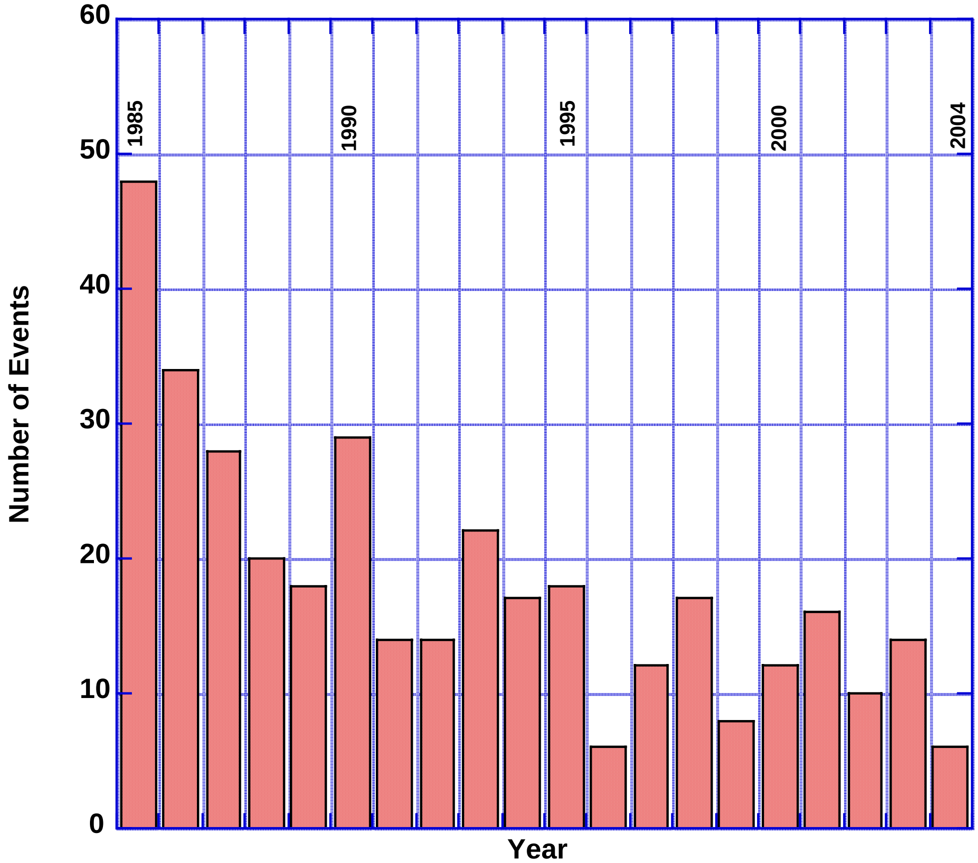

Figure 5. Number of leakage events reported per year in database. A decrease in number of leakage events from 1985 is evident. Events for 2004 includes those recorded through June.

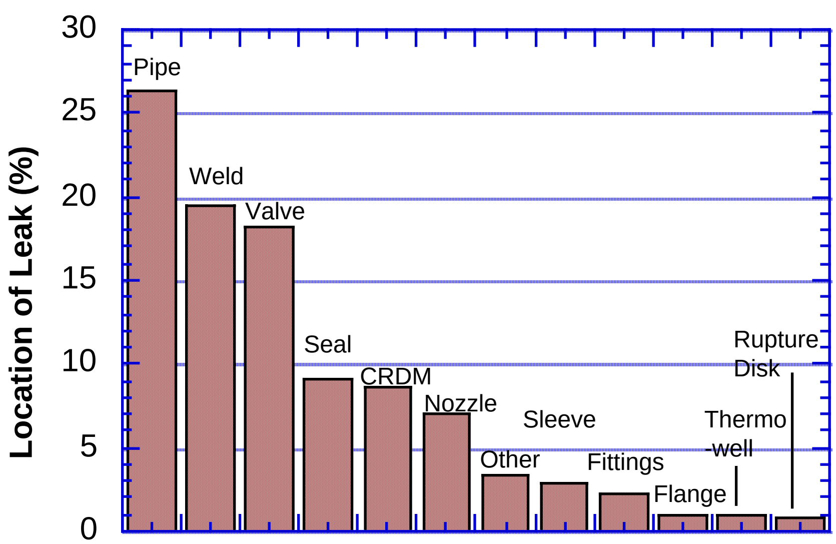

Figure 6. Number of occurrences (percentage) that a location was mentioned in a leak event report. For example, almost 20% of the time the leak involved a weld. About 8% of the leakage events reviewed involved CRDMs, usually detected through visual

detection of boric acid crystals. Cracks were involved with leakages about 40% of the time with a wide range of leakage rates [<0.00063 to 6.3 kg/s; 0.01 to >100 gpm].

In Fig. 7, the distribution of leakage rates by magnitude is shown. The number of leakages in a given range of leakage rates is given. Many leakages reported have very small leakage rates (<0.001 kg/s). They are detected visually and are reported as drips, weeping, seepage, "very small," etc. Large leakages have been detected primarily through increases in sump level, radiation alarms, inventory balance, or change in pressure.

The LERs also provide some insight into what action is taken when a leak is indicated. Frequently, the initial response from the control room operator is to initiate a surveillance test of an RCS water inventory balance.

In one case, a leakage was indicated by a radiation monitor and was accompanied by decreasing pressurizer level. An RCS water inventory balance was performed for 15 minutes with an estimated leakage rate of 0.6 kg/s (9.5 gpm). Since the leakage could not be located immediately, it was defined as unidentified leakage greater than 0.063 kg/s (1 gpm), and shutdown from full power began. The leakage was later determined to be from valve packing.

In many cases, plant procedures take action before required by TS, based on trends in unidentified leakage rate. In one example, unidentified leakage rate increased over a period of four days from 0.0063 to 0.019 kg/s (0.1 to 0.3 gpm), but was still well below the TS limit of 0.063 kg/s (1 gpm). Nevertheless, the plant was shut down from full power in accordance with procedures to confirm the source of the leakage and make repairs. A fitting thought initially to be the source of the increased leakage was found not to be the source. A crack in the above–head seal weld of the CRDM was determined to be the cause. This example indicates that leaks below 0.063 kg/s (1 gpm) can be detected with current systems in some cases [sump pumps are typically set to alarm at 0.032-0.063 kg/s (0.5–1.0 gpm)].

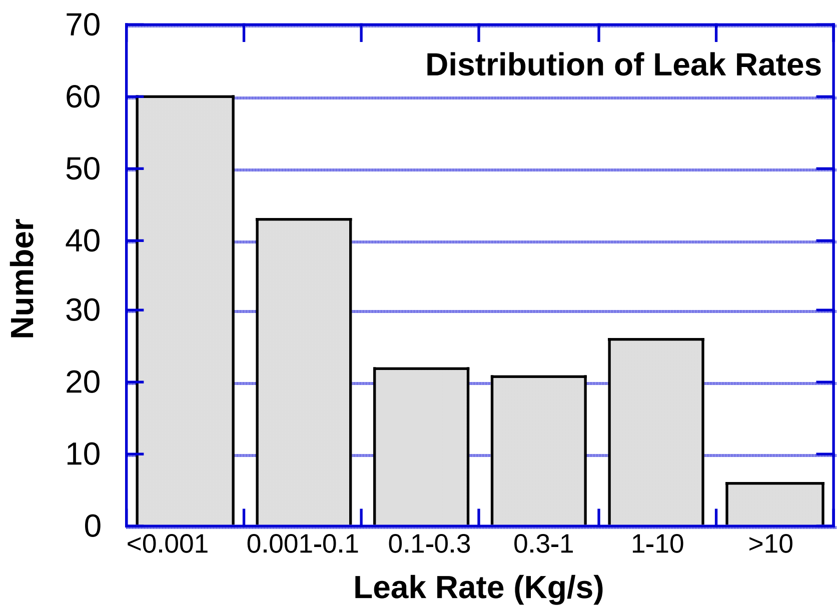

Figure 7. Distribution of leakage rates by magnitude recorded in the database beginning with 1970. The number of leaks in a given leakage rate range is indicated.

5. STRUCTURAL MARGINS FOR DIFFERENT LEAK RATES

A series of theoretical calculations were performed to determine crack lengths for various leak–rates (such as 0.006, 0.063, 0.63 or 6.3 kg/s; 0.1, 1, 10 or 100 gpm). For each pipe size the crack lengths were determined for three loading conditions corresponding to 25%, 50%, and 100% of normal operating stress at Service Level A (per ASME Section III). These should span the range of expected operating stresses, with the 50% Service Level A values representing the "typical" value.

A given leakage rate can correspond to a wide range of crack sizes. For a 0.063 kg/s (1 gpm) leak, crack lengths can range from 26 to 460 mm (1.0 to 18.1 in.) for stainless steel piping and 51 to 310 mm (2.0–12.2 in.) for carbon steel piping, depending on pipe diameter and the loading during normal operation.

The results obtained for crack lengths at various leak–rate values were compared to the critical crack length for which the specified value of the bending moment represents the maximum load to obtain a "margin of safety" against additional crack growth. Calculations giving the percent of critical crack length versus diameter for boiling water reactor (BWR) cases at 50% Service Level A stresses and leakage rates versus percent of critical crack lengths were carried out for this study. Such calculations give an estimate of the margins of safety for typical piping joints under normal operating loads.

For most piping systems, the model calculations show that the current technical specification limits on unidentified leakage provide a significant margin against gross structural failure. However, a bounding analysis of the leakage from a CRDM annulus for circumferential cracks above the J–weld shows that the typical 0.063-kg/s (1 gpm) leakage specification would not be exceeded even for cracks large enough for incipient CRDM tube failure.

Although the current requirements provide margin against gross structural failure in most cases, a 0.063 kg/s (1 gpm) leakage rate could correspond to a crack of length from 26 to 460 mm (1.0 to 18.1 in.) for stainless steel piping and 51 to 310 mm (2.0–12.2 in.) for carbon steel piping, depending on pipe diameter and the loading during normal operation.

In addition to the possibility of structural failure by crack growth, there is also the possibility that boric acid leakage could cause substantial corrosion such as occurred on the reactor vessel head at Davis-Besse. Dry boric acid results in very low corrosion rates, but with moisture present, concentrated boric acids at temperatures of 90–160°C can produce very high corrosion rates (up to 15 cm/y).

Local cooling due to leakage can create conditions for rapid corrosion by allowing aerated, concentrated boric acid solution to form on surfaces. Although the precise values of the leakage rate needed to lower metal surface temperatures to the 100–160°C range associated with high boric acid corrosion rates will depend on details of the actual geometries involved, scoping calculations (Materials Reliability Program, 2004) suggest that the critical leak rates needed to produce high corrosion rates, are of the order of 0.00063 to 0.0063 kg/s (0.01 to 0.1 gpm), well below the current TS limit. Such rates are probably also at or below the resolution limit for unidentified leakage of sump flow monitors.

6. ASSESSMENT OF LEAK MONITORING SYSTEMS

The most flexible method for detecting leakage is visual observation. The adequacy depends on the frequency of inspection and the accessibility of areas of interest. While not a principal method for leakage detection, the visual method is valuable in locating leaks. Over half of the leaks in the database had been detected visually and had very low leakage rates. Obviously, the ability to quantify a leak by observation is poor.

Field experience demonstrates that the visual method is capable of detecting leakage rates as low as 0.0006 kg/s (0.01 gpm), but the sensitivity can be even greater in some cases. An aspirin-size deposit of boric acid (approximately 400 mg) can deposit from about 0.95 L (0.25 gal) of water and could be detectable during a visual examination of reactor components. That amount of boric acid could accumulate in about a week from a leakage rate as small as 6.3×10^{-7} kg/s (10^{-5} gpm).

Humidity monitoring can detect an increase in vapor content of air resulting from a leak but suffers from a lack of quantitative information. The sensitivity could be on the order of gallons per minute when used in large volume containment areas (Instrument Society of America, 1982). Moisture sensitive tape is a continuous monitoring system in which the sensor is placed next to insulation. An electrical signal is activated when the tape becomes wet. Detection of an increase in leakage rate of 0.063 kg/s (1 gpm) within an hour is possible. Such tapes monitor a small area and have been installed at a few plants. Field experience confirms that local humidity monitors, such as the FLUS system described later, can detect leakage less than 0.1 L (0.03 gal) (Jax, 2003).

The sensitivity of temperature monitoring to detect leaks depends on volume of space, distance between sensor and leak, heat losses, normal temperature fluctuations, and presence of abnormal heat sources. Temperature monitoring probably will not detect a 0.063-kg/s (1 gpm) leak within one hour. Nevertheless, temperature sensors were installed on the relief lines in French PWRs (Shah, 1998).

A leak from the RCPB will increase the containment pressure. The consequence of having a large containment structure volume is that the leakage rate would have to be very large to be detectable by an increase in pressure. Small leakages could result in pressure variations that are in the normal range of fluctuations. Using containment pressure for leakage monitoring does not provide information on the leakage source.

Reactor coolant inventory is monitored in PWR plants but not in BWRs. This method is not particularly useful for BWRs because of the poor accuracy in detecting low RCPB leakage. Approximately 10% of all the leaks in the database were detected from inventory balance. Use of inventory balance for detecting a leakage rate of 0.063 kg/s (1 gpm) within one hour is difficult. However, under steady-state conditions, detection of 0.044 kg/s (0.7 gpm) in 2 hours and 0.021 kg/s (0.33 gpm) in 4 hours has been demonstrated under field conditions (Shah, 1998). Containment leakage, other than identified leakage which is delivered to the equipment drain sump, is drained to the containment sump as unidentified leakage.

The sump level is measured continuously by a level measuring device. A sump level and flow rate monitor can detect a 0.063 kg/s (1 gpm) leakage rate in less than 1 hour. In some plants, a 0.063 kg/s (1 gpm) leakage rate can be detected in 10 minutes (Shah, 1998). Open containment sumps collect unidentified containment leakage, including containment cooler condensate. Sump level and sump discharge flow can be monitored. Leak location is not provided. Sump pump monitors can detect a 0.063 kg/s (1 gpm) increase in leakage within one hour. Under field conditions, increases as low as a few tenths of a gpm have been detected (e.g., LER 354/1989-026-00). At Oconee, an increase in the volume of leakage on the order of 28.5 L (7.5 gal) could be detected by the sump pump (Shah, 1998), and thus a 0.063 kg/s (1 gpm) leak could, in principle, be detected in about 10 minutes. Historically, the reliability of the sump pump monitor has been good. About 10% of all the leaks in the database were reported as detected by the sump pump monitor.

Containment air cooler condensate flow runoff from the drain pans under each containment air cooler unit can be measured. A 0.063 kg/s (1 gpm) increase within one hour can be detected under normal operating conditions. This estimate is based on a calculation that shows condensate from 0.063 kg/s (1 gpm) leak rates can reach steady state in about 30 minutes (Shah, 1998).

Assuming no fuel failure, for a containment-vessel free volume of 73,700 cubic meters and a particulate activity concentration in the reactor coolant of 1.5 x 10^3 Bq/cm^3, airborne particulate monitors are capable of detecting, in principle, a 0.0063-kg/s (0.1 gpm) leak in 10 minutes (Shah, 1998). However, this type of monitor is not capable of detecting a 0.063 kg/s (1 gpm) leak within one hour under all conditions. An event occurred at Oconee 3 (Shah, 1998) where it took about 100 minutes to detect a 0.063 kg/s (1 gpm) leakage rate.

A clear understanding of the principles involved in detecting leaks from radiation monitors is necessary to avoid false alarms. For example, a decrease in reactor power level may cause an increase in the primary coolant radioactivity, and thus an apparent increase in leakage could be incorrectly surmised.

The airborne gaseous radioactivity monitor is inherently less sensitive than the particulate monitor. A leak rate of 0.13 kg/s (2 gpm) is estimated to be detectable in four hours with a gaseous monitor, assuming a coolant activity of 4 x 10^4 Bq/cm^3 of Xe-133 (Shah, 1998). With a detector sensitivity of 10^{-6} μCi/cm^3 and reactor coolant gaseous activity of 0.5 μCi/cm^3, corresponding to 0.1% fuel defects (per Southern California Edison, Ref. 12, and p. 172), 0.063 kg/s (1 gpm) leak can be detected within one hour [15]. The difficulty with the use of a gaseous radioactivity monitor arises as failed fuel is much less likely to occur, and the primary systems become less contaminated. .

Large leaks have been detected primarily through inventory balance, change in containment pressure, rise in sump level, or radiation alarms. Most leaks recorded were detected visually and were quite small. They were described as drips, weeping, seepage, “very small,” boric acid deposits, etc. The leakage rates detected by sump level changes range from 0.006 to 2.2 kg/s (0.1 to 35 gpm) with a median leakage rate of 0.095 kg/s (1.5 gpm). The median for radiation monitors (includes particulate and gas monitors) is 0.032 kg/s (0.5 gpm). Based on the reported leak rates, detecting small leakages by either a radiation monitor or sump level is possible. The median leakage rate reported for a crack is significantly less than that reported for flaws that are not cracks. Though the leakage rates reported for SCC are very low, well below the 0.063-kg/s (1 gpm) limit, non-critical flaws can be expected to leak at below the regulatory limit.

6.1 Acoustic Emission Leak Monitoring

Acoustic emission (AE) technology has the potential to provide significant improvements in leak detection capability. AE systems can provide rapid response to even small leaks, locate leaks, and monitor an entire plant. A major advantage of AE is that crack growth can be detected before the crack is through–wall due to the release of elastic energy by the growing crack. No other technique can provide this information. Furthermore, AE can be used during heat–up and pressurization when airborne monitors would not be effective. Acoustic leak detection systems can be used to monitor the entire RCS or dedicated to the monitoring of components of particular interest, such as valves.

Currently, acoustic monitoring for leakage can be carried out with a commercially available system, the Framatome–ANP “ALUS” (Kunze and Bechtold, 1995; Jax, 2003). In–service monitoring involves an array of acoustic transducers attached to the reactor coolant system or pressurizer through waveguides. Signals in the 100 to

400 kHz range are processed and the root-mean-square (RMS) values of the signal amplitude are compared with individually adjustable fixed and sliding thresholds. Typically, leakage will be detectable if the total signal is 3 dB (41%) above the background noise (Kunze and Bechtold, 1995). The estimated sensitivity varies from 0.0002 to 0.063 kg/s (0.003 to 1.0 gpm) depending on the background noise (Kunze and Bechtold, 1995; Jax, 2003). This range is similar to that reported in Kupperman, 1988. The AE sensitivity in that report was estimated to be 0.0001 to 0.063 kg/s (0.002 to 1 gpm).

The sensitivity of acoustic emission leak detection varies strongly with background noise level (in a 100–400 kHz frequency window) (Kunze and Bechtold, 1995; Jax, 2003). The lowest noise levels are in the pressurizer where the leak rate sensitivity is estimated to be as low as 0.0002 kg/s (0.003 gpm). For coolant pumps, where the noise is highest, the best observed sensitivity according to the vendor is 0.0063 kg/s (0.1 gpm). The response time is determined by the data processing time, and thus can be very short. Signal processing and decision making can be automated and controlled by computers. To calibrate the system, ultrasonic transmitters attached to the plant structure can be automatically activated during plant outages. They produce a signal with a defined intensity that simulates a leak, and attenuation measurements can then be made. Note that since the more recent development of FLUS (humidity monitor described in the following section), ALUS systems are only installed along with a FLUS system.[*]

6.2 Monitoring with Humidity Sensors

The Framatome–ANP FLUS system measures local humidity (Jax, 1994) by using a temperature– and radiation–resistant sensor tube, fabricated from a flexible metal hose with porous sintered metal elements placed at intervals of around 0.5 m. The contents of the sensor tube are pumped at fixed time intervals through a central moisture sensor that measures the absolute humidity level (the dew point) as a function of time. The location of the leak can be deduced from the time difference between the start of the pumping and the peak humidity vs. time history by using the known air velocity in the tube. The leak rate can be determined from the profile of the humidity vs. time history.

6.3 Airborne Particulate Radiation Monitor

Westinghouse has developed an airborne particulate radioactivity monitoring system (ARMS). The Westinghouse ARMS is used primarily to detect leaks in the head area of a PWR, such as CRDM canopy seals and nozzles and associated welds. Leakage in this head area releases radioactive particles, mainly Rb–88 and Cs–137, into the air volume surrounding the head. ARMS draw air from the volume through a filter to concentrate the particulates. Radiation from the collected particulates is detected by a detector consisting of a beta–sensitive plastic scintillator disk and a photomultiplier tube. The typical operation of ARMS involves collecting samples from two sources. One sample comes from the ambient atmosphere of the containment, while the other is from the reactor head. The difference of the radioactivity levels in the two samples provides a measure of leakage rate

Note that the ARMS, FLUS, and ALUS systems are not adequate for detection of very small leaks from the RPV head, such as those revealed by minor amount of boric acid crystals at the Oconee and ANO–1 Nuclear power Stations. However, for larger leaks they may be more likely to detect a significant leak prior to rupture compared to other conventional existing systems for plant monitoring.

7. SUMMARY

(1) Some currently used leakage monitoring systems have sufficient sensitivity to detect unidentified leakage of 0.032 kg/s (0.5 gpm), which is below the current 0.063 kg/s (1 gpm) limit. The analyses presented in this report confirm that the current technical specification limits on unidentified leakage can be expected to provide significant margin against structural failure of piping systems. However, leak rates well below the current limits can result in corrosion of carbon and low alloy steel components in systems containing boric acid.
(2) New leak detection technology can be used to provide greater detection sensitivity and more accurate determination of leakage locations. As part of a defense–in–depth philosophy for ensuring the integrity of the RCPB, improved leakage requirements (e.g., establishment of action requirements based on increases in unidentified leak rates, and more accurate identification, measurement, and collection of leakage from known sources to minimize

[*] Personal communication from T. Richards/W Knoblach, Framatome ANP to D. Kupperman, Argonne National Laboratory, July 5, 2004.

interference with the detection of leakage from unknown sources) could better identify RCPB breaches and prevent additional degradation of the pressure boundary.
(3) The limitations of global leakage monitoring of leakage limits for ensuring RCPB integrity should be recognized. There are portions of the RPCB for which global leakage monitoring may give very little assurance against potential loss of structural integrity. In such cases, localized leak detection systems can provide additional margin. Localized leak detection can also be sensitive enough to provide a high degree of assurance that leak rates are low enough to avoid boric acid corrosion.
(4) Significant improvements in leakage requirements will require new systems that are not only sensitive and accurate but provide the location of leaks and thus help to minimize unwanted shutdowns. Newer commercially available systems include (with vendor reported sensitivity) acoustic emission monitoring (ALUS; 0.0002 to 0.016 kg/s; 0.003 to 0.25 gpm), humidity sensors (FLUS; 0.0003 to 0.032 kg/s; 0.005 to 0.5 gpm), and air particulate detectors (ARMS; <0.006 kg/s; 0.1 gpm;). Instrumentation of the pressure vessel head with an AE system has the potential to detect leaks as small as 0.0003 kg/s (0.005 gpm). While additional technologies for leak detection, such as the use of IR spectroscopy to detect boric acid vapor in the vessel head region, may be possible, existing technologies (especially AE) already offer demonstrated capability and flexibility.
(5) The RG 1.45 may be revised to incorporate latest knowledge of state-of-the-art leakage detection and monitoring equipments, plant operating experience, and to improve the recommended leakage monitoring and measurement practice. If RG 1.45 is revised, the need for gaseous radioactivity monitoring should be re-evaluated because of better fuel performance. Consideration may also be given to include the addition of acoustic emission monitoring as an acceptable method for leak detection.

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