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Assessment of Leak Detection Systems for LWRs

Prepared by D.S. Kupperman

Argonne National Laboratory

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ABSTRACT

This topical report summarizes work performed by the Argonne National Laboratory as subcontractor on on-line leak monitoring of LWRs during the 12 months from October 1987 to September 1988.

TABLE OF CONTENTS

EXECUTIVE SUMMARY

U.S. Nuclear Regulatory Commission Guide 1.45 recommends the use of at least three different detection methods in reactors to detect leakage. Monitoring of both sump-flow and airborne particulate radioactivity is recommended. A third method can involve either monitoring of condensate flow rate from air coolers or monitoring of airborne gaseous radioactivity. Although the methods currently used by utilities for leak detection reflect the state of the art, other techniques may be developed and used. Since the recommendations of Regulatory Guide 1.45 are not mandatory, the technical specifications for operating plants have been reviewed to determine the types of leak detection methods employed. In addition, Licensee Event Report (LER) Compilations from June 1985 to March 1986 have been reviewed to help establish actual capabilities for detecting leaks and determining their source.

Generally speaking, reactor operators rely on sump pump monitoring to establish the presence of leaks, although for most reactors, the surveillance periods are too long to detect a 1-gal/min leak in 1 h, as suggested by Regulatory Guide 1.45. Also, the review of recent LERs indicates that in a number of cases, leak flow rates were above those allowed in reactor technical specifications. (The leaks reported in the LERs were primarily from valves and pumps.) It further appears from the review of LERs that radiation monitors are relatively unreliable because of high false-alarm rates.

Although current leak detection systems nevertheless appear to be adequate to ensure a leak-before-break scenario in the great majority of situations, one must also consider the possibility that large cracks may initially produce low leak rates. This situation may arise because of corrosion plugging or fouling of relatively slowly growing cracks or the relatively uniform growth of a long crack before penetration. Simply tightening the current leakage limits to improve sensitivity is not adequate, however, since this might produce an unacceptably high number of spurious shutdowns owing to the inability of current leak detection systems to identify leak sources. None of the systems currently used provides any information on leak location, and leaks must be located by visual examination after shutdown, a potentially time-consuming operation that exposes personnel to radiation. In order to improve detection of leaks through intergranular stress corrosion cracks, some U.S. utilities have installed either acoustic emission monitors or moisture-sensitive tape at specific welds.

Work at Argonne National Laboratory has demonstrated that improvements in leak detection, location, and sizing are possible with advanced acoustic leak detection technology.

ASSESSMENT OF LEAK DETECTION SYSTEMS FOR LWRs: REVISION 1

by

D. S. Kupperman

I. INTRODUCTION

It has become apparent that no currently available single leak-detection method for light-water reactors combines optimal leakage detection sensitivity, leak-locating ability, and the desired level of accuracy in leakage measurement. For example, although quantitative leakage determination is possible with condensate flow monitors, sump monitors, and primary coolant inventory balance, these methods do not provide adequate location information, and are not necessarily sensitive enough to meet regulatory-guide goals. Leak detection capability can be improved at specified sites by use of acoustic monitoring or moisture-sensitive tape (MST) [1]. However, current acoustic monitoring techniques provide no source discrimination (e.g., to distinguish between leaks from pipe cracks and valves) and no leak-rate information (a small leak may saturate the system). MST provides neither quantitative leak-rate information nor specific location information other than the location of the tape; moreover, its usefulness with "soft" insulation needs to be demonstrated.

As an indication of the concern for improving leak detection technology, we note that several investigators have evaluated the potential of acoustic techniques for detection and characterization of leaks from nuclear reactor components: Dickey et al. [2] report results for valve leakage; Collier et al. [3] report results for laboratory-grown intergranular stress corrosion cracks (IGSCCs); and McElroy et al. [4] discuss acoustic leak monitoring of nuclear reactors. Additional discussions of leak detection technology are found in references [5] and [6].

In this paper, NRC guidelines for leak detection will be reviewed, current practices described, potential safety-related problems discussed, and potential improvements in leak detection technology (with emphasis on acoustic methods) evaluated. Although information presented here is believed to be valid for most plants additional data are needed to identify exceptions. Furthermore, additional investigations will be required to adequately answer questions regarding how much improvement in leak detection reliability is possible through reactor procedural changes. It is anticipated that a more thorough report will be written in the future.

II. NRC GUIDELINES FOR LEAK DETECTION

U.S. Nuclear Regulatory Commission Guide 1.45 [7] recommends the use of at least three different detection methods in reactors to detect leakage. Monitoring of both sump-flow and airborne-particulate radioactivity is recommended. A third method can involve either monitoring of condensate flow rate from air coolers or monitoring of airborne gaseous radioactivity. Although the current methods used for leak detection reflect the state of the art, other techniques may be developed and used. Regulatory Guide 1.45 also recommends that leak rates from identified and unidentified sources be monitored separately to an accuracy of 3785 cm^3/min (1 gal/min), and that indicators and alarms for leak detection be provided in the main control room.

III. CURRENT PRACTICE: RECENT EXPERIENCES AND DEFICIENCIES

Since the recommendations of Regulatory Guide 1.45 are not mandatory, the technical specifications for 74 operating plants including PWRs and BWRs have previously been reviewed by the present authors [8] to determine the types of leak detection methods employed, the range of limiting conditions for operation, and the surveillance requirements for the leak detection systems. The results are presented again here for completeness.

All plants use at least one of the two systems specified by Regulatory Guide 1.45: All but eight use sump monitoring, and all but three use particulate monitoring. Monitoring of condensate flow rate from containment air coolers and monitoring of atmospheric gaseous radioactivity are also used in many plants.

The limit on unidentified leakage ("identified" leakage is generally that collected from monitored valves) for all PWRs is 3785 cm^3/min (1 gal/min), whereas the limit for most BWRs is 18930 cm^3/min (5 gal/min). The limits on total leakage are generally 37850 cm^3/min (10 gal/min) for PWRs and 94630 cm^3/min (25 gal/min) for BWRs. (Regulatory Guide 1.45 does not specify leakage limits, but does suggest that the leakage detection system should be able to detect a 3785-cm^3/min leak in 1 h.) In some cases, limits on rates of increase in leakage are also stated in the plant technical specifications. Two BWRs have a limit of 379 cm^3/min/h (0.1 gal/min/h); four have a limit of 1893 cm^3/min/h (0.5 gal/min/h).

Figure 1 shows a very simplified schematic for the paths of identified and unidentified leakage occurring either inside or outside the containment. Unidentified leakage ultimately passes through the sump pump unless trapped in the system (this point is discussed in a later section). In addition, condensate from the containment air cooling systems passes through a flowmeter and then the sump, adding to the unidentified leakage. Identified leakage, primarily that which is selectively collected from leaking valves, flows to a drain tank which is also pumped out. The total leakage is the unidentified and identified leakage combined. Estimates of leak rates are obtained from the cooling system flow meter, level indicators, and the frequency of operation of the pumps.

There are no requirements for monitoring leakage outside the containment. Leaks are detected by a variety of methods such as temperature and pressure rises, changes in background radiation, and visual examination during routine maintenance.

Many methods can be used to detect the presence of a leak. These include radiation monitors, sump monitors, condensate flow monitors, coolant inventory, and variations in temperature, pressure, and dew point. Generally speaking, reactor operators rely on sump pump monitoring to establish the presence of leaks. Other methods appear to be less reliable or less convenient. In most reactors, the surveillance periods are too long to detect a 3785-cm^3/min (1-gal/min) leak in 1 h, as suggested by Regulatory Guide 1.45, but it appears that this sensitivity could be achieved if monitoring procedures were modified. Simply tightening the current leakage limits to improve sensitivity is not adequate, however, since this might

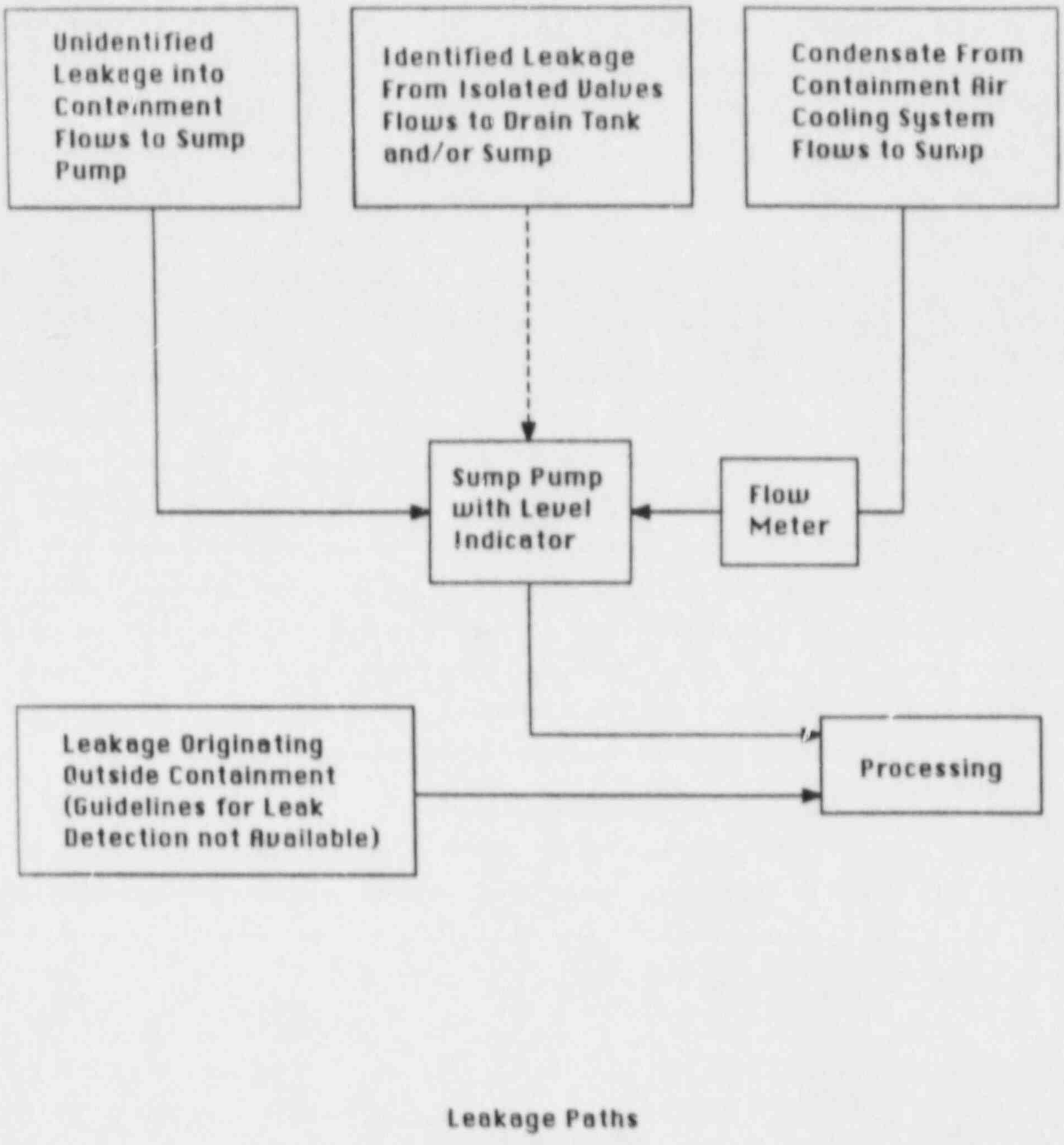

Figure 1. Simplified Schematic Representation of Flow Paths for Identified Leaks (from Monitored Valves) and Unidentified Leaks Originating Within or Outside Containment.

produce an unacceptably high number of spurious shutdowns owing to the inability of current leak detection systems to identify leak sources. None of the systems currently in use provides any information on leak location, and leaks must be located by visual examination after shutdown.

In order to help characterize more quantitatively the cause of leaks in reactors and obtain information regarding the adequacy of leak detection technology, Licensee Event Report (LER) Compilations from June 1985 to March 1988 (e.g., LER Compilation for March 1986, NUREG/CR-2000, ORNL/NSIC-200) were reviewed. These compilations contain summaries of information submitted by the nuclear power plant licensees in accordance with federal regulations. Each summary includes the date of the incident; the reactor, component, and system involved and, if a leak occurred, usually the leak rate and action taken. Out of over 4000 reported events, a total of 91 were identified as relevant to the problem of detecting leaks in the primary coolant system. PWRs account for about 70% of the reported leaks, about the same as the percentage of PWRs in the U.S. This implies that the frequency of leaks is the same for BWRs and PWRs. Pumps and valves are the main source of leaks in both types of reactor (see Table 1). Overall, there is about one false alarm for every three actual leaks. Anomalous signals from radiation monitors are the cause of these false alarms.

Differences between PWRs and BWRs with regard to leak detection have now been analyzed. The greatest differences are as follows: (a) The sump pump is reported as the detection method more frequently in BWRs than in PWRs (in 66% of BWR incidents vs. 37% of PWR incidents). (b) The radiation monitor is reported as the detection method (excluding false alarms) more frequently in PWRs. In fact, for the events studied, the radiation monitor never correctly detected a leak in a BWR (it did, however, initiate 4 BWR false alarms). Another point of interest is that inventory balance was reported as the method of detecting a leak in 16% of the PWR cases.

Tables 1-3 summarize the analysis presented above.

Table 1. Leak Sources and Detection Methods for LWRs

Leak Source	PWR + BWR (% of total)	PWR (% of total)	BWR (% of total)	PWR (% of PWR)	BWR (% of BWR)
Valves	46	35	10	48	37
Pumps	10	6	3	10	12
Small Lines	20	10	10	14	39
IGSCC	3	2	2	2	6
Misc.	21	20	2	26	6
TOTAL	100	73	27	100	100

Table 2. Leak Detection Methods for LWRs

Leak Source	PWR + BWR (% of total)	PWR (% of total)	BWR (% of total)	PWR (% of PWR)	BWR (% of BWR)
Sump Pump	46	27	19	37	66
Radiation Monitor	19	19	0	26	0
Visual Inspection	14	[illegible]	6	11	20
Inventory Balance	12	12	0	16	0
Other	10	6	4	10	14
TOTAL	100	71	29	100	100

Table 3. False Alarms Obtained with Leak Detection Systems in LWRs (% of Actual Leaks)

PWR + BWR	PWR	BWR
31	23	8

Numerous questions have arisen in connection with our attempt to assess the adequacy of leak detection. One concern was whether or not the flow path to the sump pump for unidentified leakage is unimpeded. All indications are that fluid from a leak will pass directly to the sump pump if not absorbed by the environment or insulation (see below). Levels in the containment are separated by gratings which permit the fluid to pass to the sump(s). Another concern was the time it takes to locate a leak. In general, leaks are located by visual examination, which is a slow process. (For this reason, an important benefit of improved leak location capability would be reduced personnel radiation exposure.) In addition, in the case of BWRs, the start of the examination can be delayed by up to six hours while the inert gas is removed from the drywell.

The issue of whether a significant delay in leak detection could result from the absorption of leakage by the environment or insulation was also addressed. A simple calculation based on the ideal gas law (PV = nRT) has indicated that even in the worst case, i.e., with an ambient temperature of 120°F (323 K) and the cooling condenser off, a delay of only a few hours would result from the absorption of moisture by the environment. Assuming a containment volume of 500,000 cubic feet (14,000 m^3) and a vapor pressure of 12 kPa, the maximum amount of water that can be absorbed by the air is about 300 gal. At a leak rate of 1 gal/min, saturation would be reached in about 5 hours. With the condenser on, moisture from a leak would be collected at the sump in a much shorter time. Figure 2 shows the time required to saturate the air as a function of leak rate for the worst case (cooling system off), and Fig. 3 shows the maximum volume of water that can be absorbed by the air vs. air temperature with the cooling system either off or on. The question of whether a significant amount of moisture could be held in the insulation is more difficult to answer and has not been addressed adequately at this point in the investigation.

Although sump monitoring can be reliable if conscientious surveillance is maintained, the reliability of radiation monitors is questionable, primarily for two reasons: (1) The high background radiation level in some reactors forces the alarm trip point to be set so high that the monitor is potentially insensitive to a rise in radiation level due to a leak; in one case, the radiation alarm was not activated by the presence of a 25-gal/min

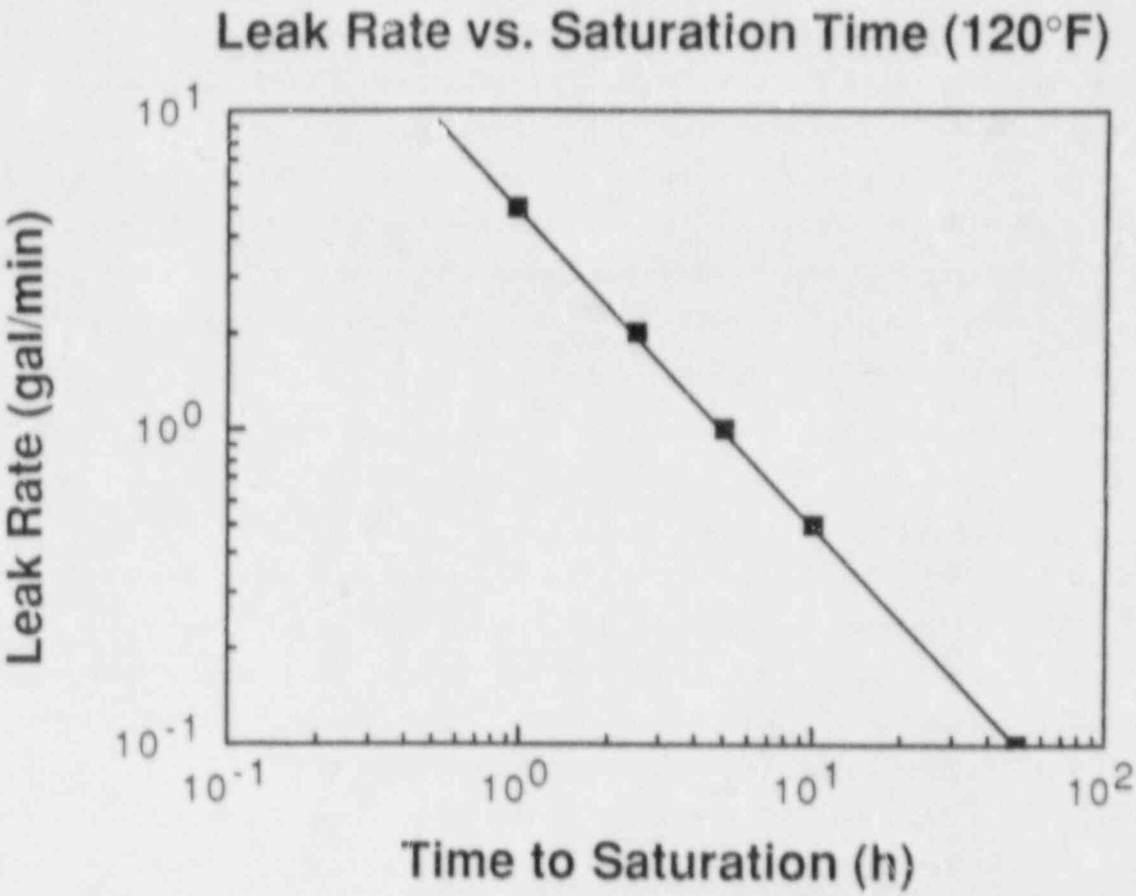

Figure 2. Plot of Approximate Maximum Time to Saturate Containment Atmosphere as a Function of Leak Rate, Based on the Assumptions that the Cooling System Condenser is Not Operating, the Containment Volume is Approximately 500,000 cu. ft., and the Ambient Temperature is 120°F.

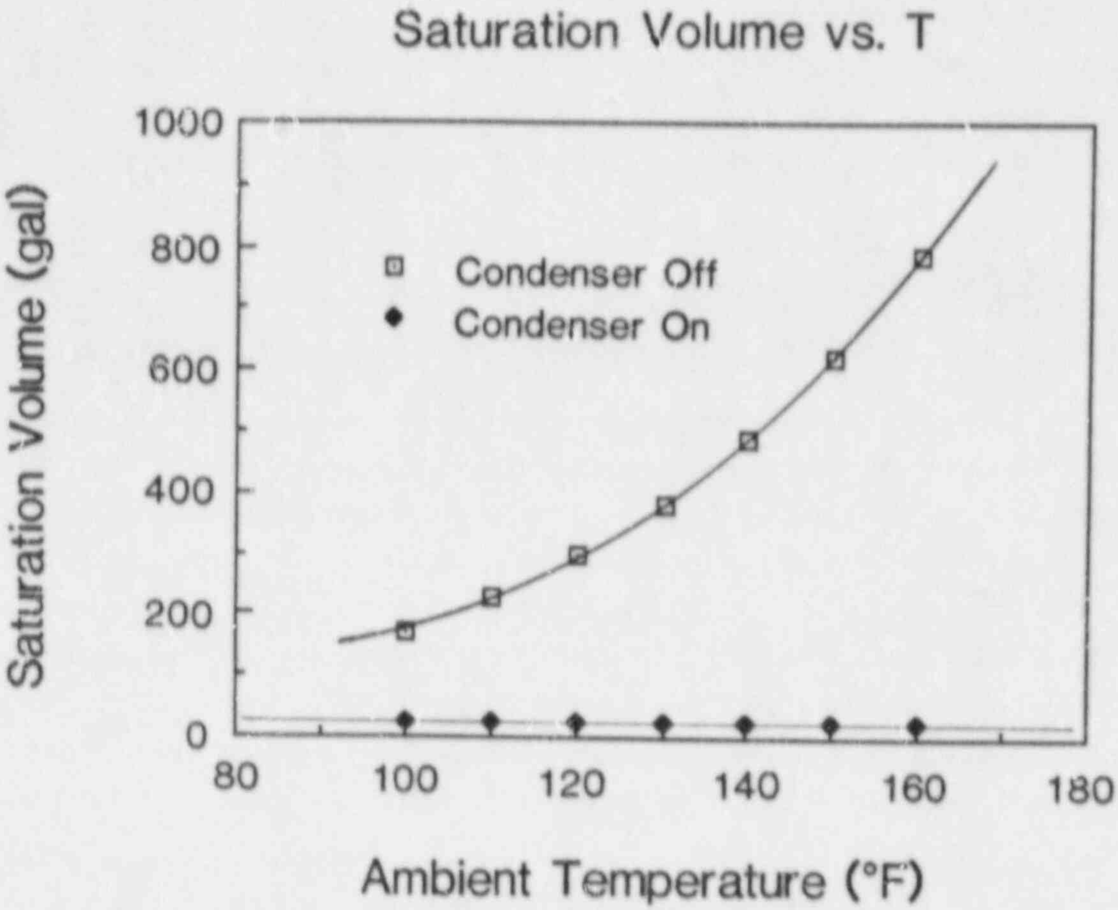

Figure 3. The Maximum Volume of Water Held as Vapor in the Containment Atmosphere (Containment Volume of 500,000 cu. ft) as a Function of Containment Temperature, With Cooling System Condenser Off or On.

leak. (2) Spurious electrical signals cause false alarms to occur at a relatively high rate.

Also addressed was the issue of whether action is taken before leaks exceed the flow rates recommended in the plant technical specifications. The answer to this question is "not necessarily," according to the LERs that were reviewed. Reported flow rates ranged from 0.3 gal/min to ">32 gal/min total"; sometimes, reports simply described leakage as "excessive."

IV. PROBLEMS ASSOCIATED WITH CURRENT LEAK DETECTION TECHNOLOGY

Although current leak detection systems are adequate to ensure a leak-before-break scenario in the great majority of situations, one must also consider the possibility that large cracks may initially produce only low leak rates. This situation could arise because of corrosion plugging or fouling of relatively slowly growing cracks or the relatively uniform growth of a long crack before penetration. In such cases, the time required for a small leak to become a significant leak or rupture could be short, depending on crack geometry, pipe loading, and transient loading (due to a seismic or water hammer event).

The shortcomings in existing leak detection systems are not simply a matter of conjecture. The Duane-Arnold safe-end cracking incident [8] indicates that the sensitivity and reliability of current leak detection systems are clearly inadequate in some cases. In the Duane-Arnold case, the plant was shut down on the basis of the operator's judgment when a leak rate of 11360 cm^3/min (3 gal/min) was detected; however, this leakage rate is below the required shutdown limit for almost all BWRs. Examination of the leaking safe-end showed that cracking had occurred essentially completely around the circumference. The crack was through-wall over about 20% of the circumference and 50-75% through-wall in the non-leaking area.

The concern about potential problems with current leak detection technology extends beyond the U.S. borders. The experience with PWRs in France has been discussed in a paper [9] presented at an international conference on surveillance of reactor coolant boundaries. French regulations related to primary coolant systems are based on NRC Regulatory Guide

1.45. In practice, however, leak detection is largely based on the chemical and control volume tank level and (for a lesser extent) the sump level and flow monitor. Locating of leakage is generally difficult and is done by local inspection after a leak is detected. The main components involved in leaks of primary coolant systems in France have been valves and, to a lesser extent, primary pump casing seals. During transient operation, the leak detection capability is reduced; as a result, the French Safety Authority has required that primary coolant leakage detection and quantification methods be improved. Otherwise, few problems have arisen in France from the primary coolant leakage detection system in the past few years.

V. IMPROVEMENTS IN LEAK DETECTION TECHNOLOGY INITIATED BY UTILITIES

In order to improve detection of leaks through IGSCCs, some U.S. utilities have installed either acoustic emission monitors (AEMs) or MST at specific welds. The AEMS have been installed at reactors in the Midwest and Southeast; MST has been installed by several other utilities. In general, these devices are installed near welds that have unrepaired crack indications or a weld overlay, and on nonconforming welds (those which have not received ultrasonic inspection because of high radiation levels or inaccessibility).

At one plant, endcap welds on a 22-in. pipe manifold have been monitored with a total of 16 MST sensors, some on the top and some on the bottom of the pipe. The system is checked during each 8-h shift to verify that the equipment is operating properly. At another reactor, MST is being used to monitor between 15 and 30 welds in the jet pump risers, the main recirculation line, and the residual heat removal (RHR) system piping. The primary concern at present is false alarms. The utility is committed to shutdown if the MST alarm goes off and the response is not confirmed to be a false alarm. During start-up, one MST sensor in the vicinity of a leaking valve triggered an alarm. This indicates adequate system sensitivity, but it also points out the need for quantitative information regarding leak characterization, location, and flow rate. In this specific case, the leak was quite large, and flow rate information was acquired through sump pump monitoring.

An AEM was installed in 1983 at a manifold sweepolet weld in a reactor in the southeastern U.S. This system includes a waveguide and commercially available components. No leak has been indicated by this AEM system, and no leaks were found during shutdown periods. This system was reproduced and tested at the Argonne National Laboratory (ANL) Acoustic Leak Detection (ALD) Facility. The analysis of the results suggests that (a) leaks as small as 7.[illegible] cm^3/min (0.002 gal/min) could be detected, (b) the acoustic background level in this particular service environment is very low, and (c) the system has limited dynamic range, saturating at 22.7 cm^3/min (0.006 gal/min).

A midwestern utility has been using AEMs on safety relief valves and has installed a similar system on a main (28-in.) recirculation line elbow. High-temperature piezoelectric accelerometers are placed directly on the pipe (one on the top and one on the bottom). The system detects signals from leaks in the 20-50 kHz range and employs a spectrum analyzer to verify that a leak is present. (Signals in a specific frequency window suggest the presence of a leak.)

Numerous low-frequency AEM systems (with high-temperature accelerometers) have been employed since 1974 to monitor valves for leakage at one eastern reactor. The primary cause of plant shutdown has been valve packing gland leaks. Leaks as small as 19 cm^3/min (0.5 gal/min) can be detected.

VI. OPPORTUNITIES FOR FURTHER IMPROVEMENTS IN LEAK DETECTION TECHNOLOGY

As suggested previously, improvements in leak detection capabilities are possible through the application of ALD devices. The effort that could lead to a field-implementable system is outlined schematically in Fig. 4. The development program would be divided into three phases. The first is the simulation of leaks and evaluation of leak detection technology under laboratory conditions. This effort would include experiments to accumulate acoustic background noise data from operating reactors. This phase would provide an estimate of ALD sensitivity and leak location and characterization capability. The second phase would be the evaluation of a breadboard system under field conditions, establishment of calibration

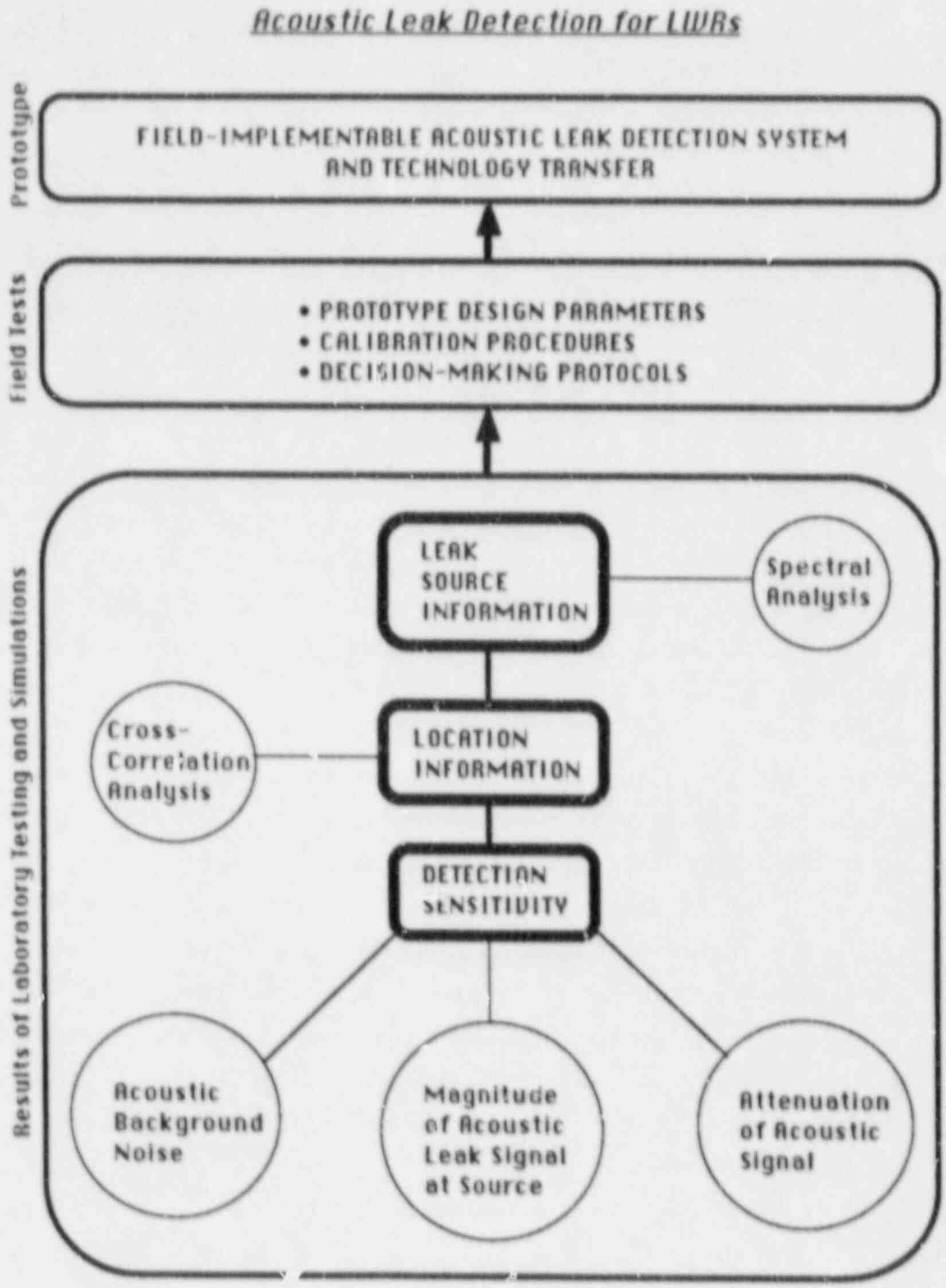

Figure 4. Schematic Representation of Proposed Phases of a Program Resulting in a Field-Implementable Acoustic Leak Detection System.

procedures, and selection of design parameters for a prototype system. The third phase would be the implementation of the prototype system and technology transfer to the utilities. A considerable amount of effort has been devoted to phases 1 and 2 through an NRC-sponsored program carried out at ANL. Detection sensitivity has been established and it has been demonstrated that cross-correlation analysis can be used to improved location capability and spectral analysis can be employed to help identify the cause of a leak. Some results of the ANL program are discussed below.

Detection of a leak by an AEM requires that $S_e = S_1 - T - N + PG > 0$, where S_e = signal excess at detector output, S_1 = source level (affected by waveguide geometry, insulation, and circumferential position), T = transmission loss down pipe, N = background noise level, and PG = system gain (all in dB). The acquisition of acoustic leak data [8], background noise estimates [10-12], and attenuation data at ANL has allowed a rough estimation of the sensitivity of an ALD system under field conditions. Figure 5 shows predicted signal-to-noise ratios (in dB) vs. distance along a 10-in. Schedule 80 pipe for three leak rates and three levels of estimated acoustic background noise. The highest level is estimated from the maximum acoustic level observed during the Watts Bar (PWR) hot functional test when the reactor was at operating temperature and pressure. The lowest level is obtained from an indirect estimate of background noise from Hatch (BWR) and the assumptions that the reactor acoustic background level will vary by a factor of 10 in the plant and that the measurement at Watts Bar was an upper-limit value. The striped area suggests possible enhancement of the acoustic signal for a 379-cm^3/min (0.1 gal/min) leak rate in a situation where the leak plume strikes the reflective insulation. Results of laboratory experiments suggest that for leak rates greater than 75.7 cm^3/min (0.02 gal/min) but less than 757 cm^3/min (0.2 gal/min), signals could be enhanced significantly, given the correct circumstances. The following equation has been used to generate the curves of Fig. 5:

$$S = 20 \log_{10} \frac{70 \quad R^{0.32}}{B} - \begin{Bmatrix} 4.5D \text{ for } D < 2 \text{ m} \\ 5.6 + 1.7D \text{ for } D \geq 2 \text{ m} \end{Bmatrix} + 6 \text{ if } 0.01 < R \leq 0.1,$$

where S is the signal-to-noise (S/N) ratio in dB, R is the leak rate in gal/min, B is the acoustic background level in μV (4, 20, or 40), and D is the distance from the leak in meters. The equation assumes a signal loss of

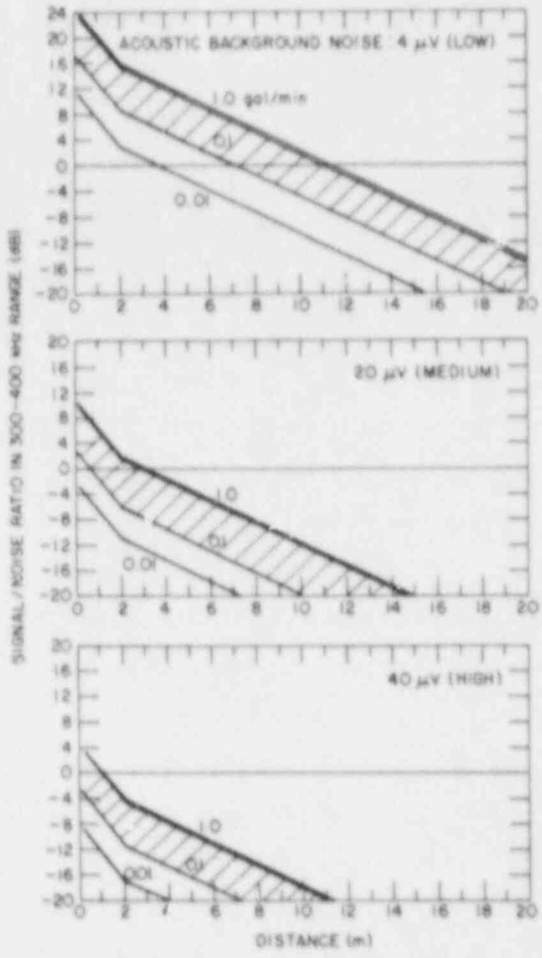

Figure 5. Predicted Acoustic Signal-to-Noise Ratios vs. Distance Along 10-in. Schedule 80 Pipe for Three Leak Rates and Three Levels of Estimated Acoustic Background Noise. The striped areas indicate possible enhancement of the signal for the 379-cm^3/min (0.1 gal/min) leak because of the presence of reflective insulation.

4.5 dB per meter for the first 2 m, followed by a further loss of 1.7 dB/m. The acoustic signal is assumed to vary as (leak rate)$^{0.32}$. A 6-dB signal enhancement has been added to each 379-cm^3 (0.1-gal/min) curve in Fig. 5 to indicate how the presence of reflective insulation could improve the signal-to-noise ratio. For low acoustic background levels, a 3785-cm^3/min (1-gal/min) leak would be detected at a distance of 11 m. With a high background level, this leak would be detected only at a distance of 1 m.

A Digital Continuous Acoustic Monitoring System (DCAMS) [1], shown in Fig. 6, has been jointly developed by GARD (a Division of Chamberlain) and ANL. Several experiments were carried out to illustrate the system capability, with encouraging results. In one, an electronic pulser and two AET 375 receivers were used to demonstrate the enhancement of location capability with demodulated acoustic signals. A program was written to allow the system operator to rectify and smooth the captured and digitized radio frequency signals. Correct location information was generated with receivers separated by ~1.5 m. In a second experiment, FAC 500-kHz broadband and AET-375 resonance receivers were attached to the ends of the pipe run (at a separation of 10 m). A continuous noise source was placed at several different locations for these tests. All tests were carried out with radio frequency signals and with the pipe empty. In each of these tests, correlograms were averaged. Unambiguous location of the source was indicated in all trials. Tests without averaging showed considerable variation in source location. In another successful test, AET-375 probes were attached to waveguides at the ends of the pipe run. Nine correlograms were averaged, but in this case the waveguides were moved slightly in a circumferential direction before each radio frequency signal was captured. This produced a spatial average and resulted in the best S/N ratio for location yet achieved.

A laboratory test has been carried out to help evaluate the capability of DCAMS to locate an actual leaking field-induced IGSCC by averaging cross-correlation functions. This averaging technique permitted a leaking field-induced IGSCC to be located, for the first time, by cross-correlation techniques.

Field trips were made to the Commonwealth Edison Co. Braidwood Nuclear

Figure 6. Photograph of Digital Continuous Acoustic Monitoring System for Enhanced Leak Detection.

Station, currently under construction, to test DCAMS under field conditions and obtain wave propagation data from electronically simulated leaks on a more extensive piping system than is available at ANL. Data from a pipe with a large gate valve between the acoustic receivers were accumulated and successfully stored in the computer. The computer system suffered no deleterious effects despite the hostile environment in which it was used. The analysis of the data indicates that cross-correlation functions can be obtained with AET-375 transducers on waveguides separated by a distance of up to 8 m without difficulty, and that averaging of correlograms can be carried out under field conditions even in the presence of a wave-distorting valve. As a result, cross-correlation analysis can be carried out even with a valve between the acoustic receivers. Thus, leak location in the vicinity of a valve is feasible.

The first step in the implementation of an acoustic leak detection and location system is to identify acoustic receiver sites and determine the spacing between waveguides required to meet the sensitivity needs of the system. The spacing scheme will depend on the type of reactor (PWR or BWR) and the level of sensitivity required. Estimates of S/N ratios for IGSCC leaks as a function of distance and acoustic background levels are presented in Fig. 5. Figure 5 can be employed as a guide for estimating the optimum sensor spacing once the desired sensitivity and background noise levels are established. Although Fig. 5 is for 10-in. pipe, the data will be assumed to be valid for all piping systems unless alternative data are available. Attenuation measurements will have to be obtained for other piping systems in the field to obtain more precise sensor spacing information. The results presented in Fig. 5 are for BWR conditions. Because of the higher pressure in a PWR, the acoustic signals for a given leak rate are higher. Adding 6 dB of S/N to the results of Fig. 5 should provide a conservative estimate of acoustic signal vs. leak rate for a PWR.

As an example, consider a BWR with 100 m of monitored piping (the approximate length of the primary pressure boundary), divided into low-, moderate-, and high-background-noise zones with lengths of 40, 40, and 20 m, respectively. For a detection sensitivity of 1 gal/min, a signal in the 300-400 kHz range, and a 3-dB S/N ratio, the required sensor spacings are approximately 10, 2, and 1 m, respectively. Therefore, 4 sensor sites are required in the 40-m low-noise zone, 20 sites in the 40-m moderate-noise zone, and 20 sites in the 20-m high-noise zone. For location analysis, three sensors are required at each site to carry out the correlation averaging routine, so altogether, 132 sensors are needed to adequately cover the reactor primary pressure boundary under the conditions proposed. For a PWR, assume 150 m of piping, divided into low-, moderate-, and high-noise zones with lengths of 60, 60, and 30 m, respectively. With an increase of 6 dB in signal intensity for a PWR compared to a BWR, Fig. 5 indicates sensor spacings of 12, 4, and 2 m, respectively, for a 3-dB S/N ratio. Approximately 105 sensors will be required [3 x (5+15+15)] to completely monitor the plant under the scenario presented. Obviously, the number of sensors can be significantly reduced if only isolated sections of the plant are monitored.

The acoustic leak detection system should be validated on leaking cracks in a laboratory facility such as the one at ANL used for this investigation. The variation in signal with leak rate, and variation in frequency spectrum with leak type, should be evaluated on the laboratory test loop. Calibration procedures could also be verified on the laboratory apparatus. Tests with field equipment must be carried out to account for differences in receivers. This can be accomplished by using a leak facility in which the field installation can be tested on IGSCC and fatigue cracks.

VII. CONCLUSIONS AND RECOMMENDATIONS

Current leak detection capabilities are not necessarily adequate to handle situations such as the IGSCC incident at Duane Arnold. Radiation monitors are potentially unreliable because of their high false-alarm rate and inherent limitations caused by high radiation background levels. Significant improvements are possible in leak detection technology, particularly with respect to leak location, and for detection of leaks during transient operation, through the use of inherently rapid acoustic leak detection systems.

Serious consideration should be given to changing condensate monitoring from an optional to a required measure in Regulatory Guide 1.45. This will provide an additional level of monitoring which, in conjunction with other techniques, will increase leak detection reliability. Efforts should be made to assist utilities with field testing and validation of existing and alternative leak detection systems, and to promote technology transfer of advanced leak detection technology. Piping Review Committee Recommendation A5 (Vol. 3), "Validation of the Reliability of Leak Detection Systems," should be implemented. [13]

ACKNOWLEDGMENTS

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APPENDIX

SUMMARY OF LICENSEE EVENT REPORTS

	Component	System	Event	Leak Rate
1	Isolation Valve	Primary	Valve disk-to-seat leakage during performance of LLRT.	
2	Socket Weld	Recirculation Discharge Bonnet Vent Valve	Leak from temporary hose and small vibration-induced fatigue crack in socket weld.	Exceeded tech. spec.
3	0.75-in. Line	Water Cleanup System	Unidentified drywell leakage from vibration-induced weld crack in test line.	< 1 gpm
4		Lower Containment	Radiation alarm caused containment isolation.	False alarm
5	Steam Line Drain Valve	Main Steam Line Header	Steam leak discovered on a main steam line drain valve.	?
6	Recirculation Pump	Recirculation System	Excessive total leakage from drywell floor drain caused by leaking pump shaft seal. Reactor shut down when leakage reached 32 gpm.	> 32 gpm
7	Drain Lines	Primary System Pressure Boundary	Reactor shut down as a result of leakage in two 0.75-in. drain lines. Failure mechanism was high-cycle fatigue.	?
8	Pump Control Valve	Centrifugal Charging Pump	Sudden leak from packing of charging pump level control valve.	10 gpm
9	Coolant Pump Seal	Reactor Coolant Isolation Valve	Leak in reactor coolant pump seal isolation valve caused by blown packing.	about 5 gpm

Detection Method	Evaluation and Action	Reactor
?	Repaired.	Fitzpatrick BWR
Drywell leakage	Valve and pipe removed and socket plugged and welded. Procedure revised.	Browns Ferry 1 BWR during startup
Drywell leakage	Pipe replaced and hanger clamp installed.	Browns Ferry 3 BWR during operation
Radiation level exceeded threshold (>2 times normal)	Procedure modified.	Cook 1 PWR during reactor power increase
Routine visual inspection	Valve isolated for repair at next cold shutdown.	Connecticut Yankee PWR during operation
Drywell drain	Pump rebuilt and reinstalled	Hatch 1 BWR during steady-state operation
Routine inspection	Lines repaired; other drain lines examined. Need for additional supports assessed.	WPPSS 2 BWR during normal operation
Elevated radiation level	Packing rings replaced; packing ring repair procedure changed.	Zion 2 PWR during normal operation
Sump pump level	Plant shut down, valve repaired.	Beaver Valley 1 PWR during normal operation

	Component	System	Event	Leak Rate
10	Diaphragm Valves	CVCS Letdown Flowpath of Reactor Coolant System	Leakage (from diaphragm valves) exceeded maximum allowed value of 1 gpm.	> 1 gpm
11	Pipe	Feedwater Heater Normal Level Control Valve	Steam leak of unknown origin caused by rupture of pipe downstream of feedwater valve.	?
12	Stem Packing of Pressurizer Spray Valve	Containment	Excessive leakage in RCS resulting from failure of the valve stem packing of a pressurizer spray valve.	3 gpm of which 1.7 gpm was unidentified
13	2-in. Instrument Line	Containment	Reactor shut down when high water level alarm tripped. Leak was from crack in fillet weld in 2-in. instrument line above active fuel and 13 in. outside the reactor vessel dry well.	0.3 gpm
14	1-in. Vent Line	Steam	Reactor shut down because of unidentified leakage in reactor coolant ystem. Leak was from crack in vent line. Probable cause of failure was fatigue crack due to inadequately supported pipe.	17 gpm
15	Vent Line	Primary Coolant System Loop #1 Water Box	Occasional drip of borated water from insulation was discovered during inspection of containment. A pinhole IGSCC leak was found.	"Small"
16	Pipe Plug	Containment Fan Coil Unit Motor Cooler	Leakage into sump.	Up to 1.3 gpm

Detection Method	Evaluation and Action	Reactor
Sump pump level (?)	Diaphragm valves tightened.	Byron 1 PWR
?	Pump shut down and isolated; eroded pipe replaced	Connecticut Yankee PWR at 100% power
Water inventory balance and air-borne radiation level	Packing repaired and valve system modified.	San Onofre 3 PWR
Sump monitor	Consultation held with GE.	Pilgrim BWR during normal operation
Drain accumulator tank dump alarm shutdown	Task force formed to evaluate problem.	Rancho Seco PWR
Visual	Vent line replaced.	[illegible]kee Rowe PWR
Sump monitor	Pipe plug tightened.	Salem 1 PWR

Component	System	Event	Leak Rate
17	RWCU Isolation	RWCU system isolated as a result of spurious signal in leak detection system.	False alarm
18 Riser Weld	Recirculation System Header to Riser	1-in.-long through-wall IGSCC in HAZ detected after IHSI.	-
19 Socket Weld	Between isolation valve and pipe spool piece of the 2-in. reactor drain line	Reactor shut down because of leak in a porous socket weld.	"Small"
20 Motor Cooler Head Gasket	CFCU Motor	Sump pump run indicated unidentified containment leakage greater than 1 gpm and classified as RCS leakage.	About 1 gpm
21	Containment Radiation Monitor	Radiation alarm as a result of a spurious electrical noise spike.	False alarm
22	Containment	Airborne-particulate radiation monitor failed as the result of a failed power supply bridge diode.	False alarm
23	Containment	Spurious high alarms from airborne-particulate radiation monitor.	False alarm
24	Containment	High radiation level indicated by gaseous radiation monitor.	False alarm
25	Containment Ventilation System	Containment isolation caused by spurious spike in gaseous radiation monitor.	False alarm

Detection Method	Evaluation and Action	Reactor
System trip	Extensive testing planned.	Duane Arnold BWR: normal operation
?	Weld overlay repaired and penetrant tested.	Browns Ferry 2 BWR following IHSI
?	Spool piece (including weld) replaced.	Pilgrim 1 BWR during normal operation
Sump pump monitor	Motor cooler replaced.	Salem 1 PWR during normal operation
Radiation monitor	Cause of spurious signal investigated.	San Onofre 2 PWR
Instrument failure signal	Diode replaced.	San Onofre 3 PWR with system at full power
Airborne-particulate radiation monitor	Computer chip replaced.	Cook 1 PWR during hot standby
Gaseous radiation monitor		Catawaba 2 PWR
Gaseous radiation monitor	Procedure modified. (No obvious cause for false alarm found.)	Diablo Canyon 1 PWR during power operation

	Component	System	Event	Leak Rate
26		Containment	High level of gaseous radiation monitor tripped on high iodine signal and activated the containment purge isolation system.	False alarm
27	Bypass Manifold Flow Indicator	Lower Containment	Alarms received from radiation monitors. A small leak was discovered at a loose vent plug of the Bypass Manifold Flow Indicator.	?
28	Recirculation Pump Discharge Bypass Valve	Reactor Coolant System	Reactor shut down because of excessive unidentified drywell leakage. Leak caused by severe leak in bypass valve.	10 gpm
29	O-Ring in Reactor Vessel	Reactor Coolant Pressure Vessel	Reactor shut down because of leak detected during hot standby; caused by sealing failure of an outer o-ring between vessel and head flange.	2-3 gpm
30		Containment	False indication from gaseous radiation monitor caused containment ventilation isolation. One of six events of the same type caused by spurious signals. Cause unknown.	False alarm
31		Containment	Spurious signal tripped radiation monitor and led to containment purge isolation.	False alarm
32	2-in. Valve	Cold Leg Loop Stop Valve Bypass Line Isolation Valve off Loop 'A'	Unidentified reactor coolant system leakage increased to level above tech. spec. Reactor placed on hot standby while a packing leak in bypass line valve was fixed.	> 1 gpm

Detection Method	Evaluation and Action	Reactor
Gaseous radiation monitor	Charcoal filter changed.	San Onofre 3 PWR at 100% power
Radiation monitor	System repaired.	Cook 2 PWR
Sump monitor	Valve repaired.	Susquehanna 2 BWR during normal operation
?	O-ring repaired; use of different type considered.	Turkey Point 3 PWR during hot standby
Gaseous radiation monitor	Cable connectors replaced; problem investigated	Diablo Canyon 1 PWR during power operation
Radiation monitor	Cause unknown. Monitor worked properly after event.	Palo Verde 1 PWR during normal operation
Increase in sump flow and containment radioactivity level	Maintenance carried out to reduce leak rate to acceptable level.	North Anna 2 PWR during normal operation

	Component	System	Event	Leak Rate
33	Instrument Isolation Valve	Containment	Reactor scram occurred during an instrument surveillance because of a leak in an isolation valve seat.	?
34	Valve Packing Glands	Reactor Coolant System	Reactor shut down because of leak through packing glands on pressurizer valve loop seal drain valve and pressurizer instrumentation tap.	1.0+ gpm
35	Cracked Pipe	Cooling Water Inlet Pipe to Fan-Coil Unit	Leakage from 1-in.-long circumferential crack in $2\frac{1}{2}$-in.-diameter pipe, probably the result of fatigue.	?
36		Containment	Radiation monitors began responding to high activity in containment due to failed cladding on fuel rods.	False alarm
37	Valve Packing	Resistance Temperature Detector Bypass Loop Valve	Airborne-particulate radiation monitors indicated a leak which could not be found during containment inspection. A packing valve leak was found during walkdown and repaired.	?
38	Fan Cooler Tube	Containment	Leakage into containment exceeded tech. spec. Leak was found in fan cooler.	> 10 gpm
39	Bypass Valve	Reactor Coolant System	Leakage noticed in containment and found to be coming from an RCS Bypass Line Valve. Reactor placed in hot standby and valve fixed.	> 10 gpm

Detection Method	Evaluation and Action	Reactor
Indication of low reactor water level	?	WPPSS 2 BWR
Inventory balance surveillance	Packing adjusted to bring leak rate below 1 gpm.	Salem 2 PWR during operation
Sump pump monitor	Temporary patch applied.	Prarie Island 1 PWR during power operation
Gaseous radiation monitor	Trip of monitor classified as an administrative deficiency and monitor classified as failed component.	Catawba 1 PWR at 100% power
Airborne-particulate radiation monitor	Valve packing repaired.	Cook 2 PWR during hot standby
Sump pump monitor	Fan coil unit replaced.	Indian Point 2 PWR
Increased makeup flow and increased pumping frequency of primary drain transfer tank		North Anna PWR at full power

Component	System	Event	Leak Rate
40	Lower Containment	Airborne-particulate monitor indicated excessive level during increase in temperature and pressure.	False alarm
41 Unknown	Drywell	Unidentified leakage into drywell floor drain exceeded 5-gpm limit. Reactor mode switch was placed in shutdown position and leakage was reduced. Cause unknown.	> 5 gpm
42 Valve	Primary Coolant System (PCS)	Excessive unidentified PCS leakage was identified. Containment entry revealed a leak from packing on an RHR valve.	"Excessive"
43 Coolant Pump Flange	Reactor Coolant System	Airborne-particulate radiaton monitor indicated high level of activity. During containment closeout tour, a leaking coolant-pump flange was discovered.	?
44 Valves	Reactor Coolant ystem	RWCU DIV 1 isolation valves isolated because of leak through 2 air-operated valves.	10-15 gpm
45	RWCU System	Four isolations of the RWCU occurred during heatup. Inspection of the area of alarm revealed no steam leakage.	False alarm
46 Valve	Pressurizer Liquid Sample Line of RCS	Reactor placed in hot shutdown to repair leaking valve.	3-4 gpm

Detection Method	Evaluation and Action	Reactor
Airborne-particulate monitor	Set points of monitor set higher to reflect actual background radiation levels in the containment.	Cook 1 PWR
Sump monitor	Cause being evaluated.	Hatch 1 BWR during normal operation
Coolant system leak rate calculation	Temporary repair made; permanent repair planned.	Palisades PWR at 98% power
Airborne-particulate radiation monitor	None reported.	Cook 1 PWR during hot standby
High differential flow alarm	Valves repaired.	Riverbend 1 BWR during operation
Leak detection temperature monitors	Temperature monitor set points were conservatively low and were reset.	WPPSS 2 BWR
?	Valve shut and repacked.	Rancho Seco PWR at 97% power

	Component	System	Event	Leak Rate
47	Valve	Spray By-pass Valve	Excessive leakage of the spray bypass valve was detected while returning to operation.	3-4 gpm
48	Compression Tube Fittng	NC Flow Transmitter	Reactor shut down: 1-gpm unidentified RCS leakage.	> 1 gpm
49	Several Valve Stems and RC Pump Main Flange	Reactor Coolant System	Reactor shut down because of unidentified leakage greater than 1 gpm. Leaks were from many sources.	1.3 gpm
50	Three Valves	Reactor Coolant System	Reactor shut down because of unidentified leak in excess of 1 gpm. Leaks were from several valves.	> 1 gpm
51	Upper Conoseal on In-Core Thermocouple Support Column, Reactor Vessel Seal Ring, and Cracked Pipe on Drain Valve Connection	Reactor Coolant System	Reactor shut down because of excessive unidentified leakage. Unit placed on line after conoseal, reactor vessel seal rings and pipe on drain valve connection replaced.	1.28 gpm
52	Instrument Valve	Reactor Coolant System	Reactor placed in hot standby because of excessive unidentified leakage. Leak was determined to be from instrument root valve packing gland failure.	> 1 gpm
53	Socket Weld	Reactor Coolant System	Water leaked through a cracked socket weld for 40 min at 25 gpm. THE RADIATION LEVEL DID NOT INCREASE.	25 gpm

Detection Method	Evaluation and Action	Reactor
?	Valve repaired; local leak rate testing carried out.	Wolf Creek PWR
Not reported	Compression tube fitting tightened.	Catawba 1 PWR at 100% power
Not reported	Leaking valves repaired and flange bolts tightened.	Catawba 1 PWR at 100% power
Not reported	Visual examination revealed 3 leaking valves. Packing adjusted on first and replaced in second; seal ring replaced on the third.	Millstone 2 PWR at 100% power
Leak rate test	Unit placed on-line after cono-seal, reactor vessel seal rings and pipe on drain valve connection replaced. Leak rate reduced to 0.34 gpm.	North Anna 1 PWR
Not reported	Affected flow transmitter isolated. Replacement of valve planned.	Oconee 2 PWR
Not reported	Section of piping cut out and sent to Westinghouse for failure analysis.	Catawba 1 PWR

	Component	System	Event	Leak Rate
54	?	Primary Coolant System	?	3.8 gpm
55	Steam Line Elbow	Secondary Side of PWR	Reactor shut down because of break in 6-in. steam line elbow near condenser on secondary side.	Through 4-in. break
56	Pump Seal	Recirculation System	Reactor shut down because of leak in recirculation pump.	Exceeds Tech. Spec. Limit
57	Head Spray Piping		Leak found in 4-in. head spray piping.	?
58	2-in. by 1-in. Reducing Coupling	Reactor Water Level Instrument Line	Reactor shut down because of leak caused by a cracked weld in a reducing coupling which joined reactor water level instrument to vessel penetration.	Increase in unidentified drywell leakage
59	Coolant Pump	Reactor Coolant System	Excessive unidentified leakage indicated; during containment walkthrough, it was traced to packing leak and leaking diaphragm.	14.35-16.82 gpm total, including 0.75-2.48 gpm unidentified
60	Valves	Lower Containment	Airborne-particulate radiation monitor tripped. Leaks through packing from several valves were detected by visual examination.	Not indicated
61	Valves	Reactor Coolant System	Excessive reactor coolant leakage from 2 valves.	> 1 gpm
62	RHR Letdown Line	Residual Heat Removal System	Broken weld on RHR system letdown line was discovered.	3000-7000 gal total
63	Relief Valve and Reactor Head Vent System Valves	Primary Coolant System	Unidentified leakage detected and reactor placed in hot shutdown.	1.25 gpm

Detection Method	Evaluation and Action	Reactor
?	Problem under investigation.	Cooper BWR
Operator heard loud rushing noise from generator. Leak was then located by visual examination	Steam line repaired; condition of similar piping checked.	Ginna PWR
?		Grand Gulf 1 BWR
Visual detection during plant tour	Head spray line was cut and capped.	Pilgrim 1 BWR
Sump level increase	Crack caused by incomplete root pass penetration plus thermal stress. Area repaired and sleeved.	Pilgrim 1 BWR
Not specified	Backseating procedure reduced leakage to acceptable levels.	Surry 2 PWR
Airborne-particulate radiation monitor	Alarm set points readjusted to better reflect the background radiation levels.	Cook 2 PWR
Visual examination	Valves repaired.	McGuire 2 PWR
	Caused by unusual service condition and loose packing leading to water hammer.	McGuire 2 PWR
Not reported	Valves repaired and returned to service.	Palisades PWR at 98% power

	Component	System	Event	Leak Rate
64		Containment	Alarm on upper containment normal range area monitor tripped.	False alarm
65	Main Steam Line	Primary Containment	With system at 15% power, containment isolation and scram was initiated due to a main steam line high flow signal. No leak or problem was found.	False alarm
66		Containment	Inboard containment isolation valves for RWCU system automatically isolated due to faulty high differential leak detection temperature switch.	False alarm
67	Valves	Containment	Airborne-particulate radiation monitor alarms resulted in reactor trip. Entry into containment revealed several leaking valves.	Not reported
68	Pump Suction Valve	Recirculation System	Unidentified leakage into drywell drain exceeded 5-gpm limit.	> 5 gpm
69	Impulse Line Compression Fittings	PZR Impulse Lines	High makeup rate to the volume control tank indicated unidentified reactor coolant leakage.	9 gpm
70	Expansion Joint	Service Water Return Line from Recirculation Spray Heat Exchanger	Service water leak discovered and plant rampdown initiated.	1 gpm
71	Packing on Pressurizer Spray Bypass Valve	Reactor Coolant System	Unidentified RCS leakage exceeded tech. spec. limit of 1 gpm.	7 gpm

Detection Method	Evaluation and Action	Reactor
Radiation monitor	Exact cause not determined; electronics modified.	Cook 2 PWR
Main steam line high flow signal	Cause not determined.	La Salle 2 BWR
Temperature switch	Temperature switch modules replaced.	Perry 1 BWR
Airborne-particulate radiation monitor	Valves repaired during shut down.	Cook 1 PWR
Sump pump	Valves repaired.	Hatch 1 BWR at 97% power
Volume control tank level	Fittings replaced.	McGuire 1 PWR
Not reported	Expansion joint replaced.	Surry 2 PWR at 100% power
Radiation level	Packing repaired during cold shut down.	Wolf Creek 1 PWR

	Component	System	Event	Leak Rate
72	Rupture Disc on Pressurizer Relief Tank	Containment Pressurized Relief Tank	Sump pump experienced excessive pump runs. Containment entry revealed leak from pressurizer relief tank. Leak was from valve packing.	20 gpm
73	Valve (suspected)	Drywell	Unidentified leakage reached 5 gpm.	5 gpm
74	Pressurizer Relief Tank			
75	Impulse Line Compression Fittings	PZR Impulse Lines	High makeup rate to the volume control tank indicated unidentified reactor coolant leakage.	9 gpm
76	Expansion Joint	Service water Return Line from Recirculation Spray Heat Exchanger	Service water leak discovered and plant rampdown initiated.	1 gpm
77	Packing on Pressurizer Spray Bypass Valve	Reactor Coolant System	RCS unidentified leakage exceeded tech. spec. limit of 1 gpm.	7 gpm
78	Rupture Disc on Pressurizer Relief Tank	Containment Pressurizer Relief Tank	Sump pump experienced excessive pump runs. Containment entry entry revealed leak from pressurizer relief tank. Source of leak was packing from valve.	20 gpm

Detection Method	Evaluation and Action	Reactor
Sump monitor		Zion 1 PWR
Sump monitor	Reactor shutdown	Limerick 1 BWR
Volume control tank level	Fittings replaced	McGuire 1 PWR
Not reported	Expansion joint replaced	Surry 2 PWR at 100% power
Increase in radiation levels in containment. Water inventory balance was used to establish leak rate.	Packing repaired during cold shutdown	Wolf Creek 1
Sump monitor	--	Zion 1 PWR

	Component	System	Event	Leak Rate
79	1C RTD Bypass	1C RTD Bypass	Leak discovered.	6 gpm
80	Check Valve	Residual Heat Loop B	Slowly increasing drywell leakage (several days) led to shutdown from 80% full power. Packing leak found after drywell entry.	5.04 gpm (unspecified)
81	Three Primary System Valves and Reactor Coolant Pump Seal Housings	Reactor Coolant System	Excessive leakage caused shutdown from 100% power.	> 1 gpm
82	RTD Bypass Flow Element	Reactor Coolant System	Leakage exceeded tech. spec. Reactor shutdown from 100% power.	> 1 gpm
83	Pump	Reactor Water Cleanup	RWCU system isolated on a signal from differential temperature switch. Mechanical seal on RWCU pump was leaking.	about 5 gpm
84	Valves	Residual Heat Removal	Reactor shutdown after unidentified leak rate exceeded 5 gpm.	8-10.5 gpm
85	False Alarm	Reactor Water Cleanup	Spurious trip of the RWCU LD temperature system.	False alarm
86	False Alarm	Process Radiation Monitor	Airborne radioactivity monitor tripped during 100% power operation. Failure due to faulty transistor.	False alarm

Detection Method	Evaluation and Action	Reactor
Not specified though 17.5 hours were required to locate leak after the unit was offline.	Shutdown	Catawba 1
Sump monitor	Valve repacked	Susquehanna 2 BWR
Sump monitor	Repair	North Anna 2 PWR
Sump Monitor		North Anna 1 PWR
Differential temperature	Seal replaced	Arnold BWR
Sump monitor	Packing of valves replaced.	Riverbend 1 BWR
Differential temperature signal		LaSalle 1 BWR
Gaseous radiation monitor	Instrument repaired	Turkey Point 4 PWR

	Component	System	Event	Leak Rate
87	Valve	Reactor Coolant	Unidentified leak rate exceeded tech. specs. Leak was found in valve stem packing of an NC power-operated relief valve isolation valve.	> 1 gpm (unidentified)
88	Pressurizer	Primary System	Unidentified leak > 1 gpm detected. Visual inspection established leak to be at top of pressurizer.	2.4 increasing to 8 gpm
89	Valve	Reactor Coolant System	Unidentified leakage exceeded 1 gpm and reactor was shut down. Leak was found to be from packing of a letdown isolation valve.	1.2 gpm max
90	Thermal Barrier Flange	Pressure Boundary	Leak in pressure boundary resulted in controlled shutdown.	?
91	Loop Stop Valve	Reactor Coolant System	Unidentified leakage of 47 gpm resulted in shutdown. Cause was failed packing on loop stop valve.	47 gpm

Detection Method	Evaluation and Action	Reactor
Sump monitor	Action not specified	Catawba 2 PWR at 100% power
?	Took over 12 hours to locate by visual inspection after entering containment	Robinson 2 PWR
Sump monitor	Leak source repaired.	Cook 1 PWR
?	Crack found in RCP thermal barrier flange in the area of the seal injection inlet nozzle originating in the weld root.	Summer 1 PWR
Sump monitor and visual	Loop stop valve replaced.	Surry 1 PWR 100% power

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