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# Aging Management and Performance of Stainless Steel Bellows in Nuclear Power Plants

Sandia National Laboratories

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# Aging Management and Performance of Stainless Steel Bellows in Nuclear Power Plants

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Prepared by
J.L. Cherry, J.M. Clauss, M. Pilch, J.J. Gregory, J.A. Smith

Sandia National Laboratories
Albuquerque, NM 87185-0744

H.L. Graves III, NRC Project Manager

**Division of Engineering Technology**
**Office of Nuclear Regulatory Research**
**U.S. Nuclear Regulatory Commission**
**Washington, DC 20555-0001**
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# Abstract

In commercial nuclear power plants, many containments use stainless steel bellows at piping penetrations, in drywell-to-wetwell vent lines, or as a part of the fuel transfer tube. Bellows are an integral part of the primary containment pressure boundary in nuclear power plants. These bellows allow for thermal expansion and contraction of the pipe, vent line, or fuel transfer tube, and accommodate relative motion between the containment and other structures, such as a shield wall.

Ensuring that bellows in operational plants remain leak tight is an important safety issue. A number of bellows have been replaced when transgranular stress corrosion cracking caused them to leak. Determining how to evaluate bellows for age-related damage is a critical issue that must be addressed for plants operating under their original license, as well as for plants applying for license extension.

The types of bellows, locations in a containment, and required bellows leak tests are described for major categories of boiling-water reactor (BWR) and pressurized-water reactor (PWR) power plants. Compilation of instances of bellows cracking, excessive leakage, or structural integrity questions, as well as testing issues and other degradation problems, are reviewed and summarized. The aging mechanisms that caused the degradation are discussed. Inspection methods currently in use, and a typical industry approach to managing bellows aging are described. Degradation that resulted from corrosion, misalignment, and other damage was considered. The risk and regulatory significance of containment bellows when subjected to Current License Basis (CLB), degraded, and severe accident (SA) conditions are also addressed.

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# Acknowledgments

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Information about penetration failures was gathered by Sandia National Laboratories (SNL) for the "Aging Management Guideline for Commercial Nuclear Power Plants – Containment Penetrations," which is an unpublished work prepared for the US Department of Energy's (DOE's) Commercial Light-Water Reactor program. The DOE program, with Dennis Harrison as the DOE Nuclear Energy program manager, will publish their report through the Electric Power Research Institute.

This report uses reference and database information gathered for the DOE program. The DOE report is broad and considers all penetrations, while this report specifically addresses bellows issues. This project's additional research led to valuable insights that apply specifically to the bellows in nuclear power plants.

# Acronyms and Initialisms

| | |
|---|---|
| AMR | aging management review |
| ASME | American Society of Mechanical Engineers |
| B&PV | Boiler and Pressure Vessel |
| BWR | boiling-water reactor |
| CFR | Code of Federal Regulations |
| CLB | Current License Basis |
| CLWR | commercial light-water reactor |
| DBA | design basis accident |
| DOE | Department of Energy |
| ECL | early containment leak |
| EJMA | Expansion Joint Manufacturers Association |
| EPRI | Electric Power Research Institute |
| FR | Federal Register |
| FSAR | Final Safety Analysis Report |
| ILRT | integrated leak rate test |
| INPO | Institute for Nuclear Power Operations |
| IPA | Integrated Plant Assessment |
| ISI | in-service inspection |
| LCL | late containment leak |
| LER | License Event Report |
| LLRT | local leak rate test |
| LOCA | loss of coolant accident |
| MELCOR | system level code |
| MIC | Microbiologically Influenced Corrosion |
| MPFF | maintenance preventable function failure |
| NCF | no containment failure |
| NEI | Nuclear Energy Institute |
| NPRDS | Nuclear Plant Reliability Data System |
| NRC | Nuclear Regulatory Commission |
| ORNL | Oak Ridge National Laboratory |
| PB | Peach Bottom |
| PRA | probabilistic risk assessment |
| PT | Penetrant Test |
| PWR | pressurized-water reactor |
| RC | reactor coolant |
| RCIC | reactor core isolation cooling |
| RHR | residual heat removal |
| RPV | reactor pressure vessel |
| RWCU | reactor water cleanup |

| | |
|---|---|
| SA | severe accident |
| SCC | stress corrosion cracking |
| SCFH | standard cubic feet per hour |
| SEQ | Sequoyah |
| SG | steam generator |
| SNL | Sandia National Laboratories |
| SSCs | systems, structures, and components |
| SSE | safe shutdown earthquake |
| TF | trichlorotrifluoroethane |
| TGSCC | transgranular stress corrosion cracking |
| TIP | traversing in-core probe |
| TLAA | time-limited aging analyses |
| TR B | test return, train B |

# 1. INTRODUCTION

## 1.1 Purpose and Objective

Most steel and a few concrete containments are constructed with stainless steel bellows at piping penetrations that are subject to large movements. Bellows allow relative movement between the steel containment shell and the concrete shield wall, while minimizing the load imposed on piping and the wall. Bellows are also used in other locations, such as boiling-water reactor (BWR) Mark I containment drywell-to-wetwell vent lines. Pressurized-water reactor (PWR) and BWR Mark III containments may include fuel transfer tube bellows, which allow relative movement between the containment and the transfer tube to the spent fuel pool. Bellows are an integral part of the primary containment pressure boundary in nuclear power plants. If an accident occurs, bellows may be subjected to pressure and temperature conditions that cause considerable axial and lateral deflections.

To ensure public safety, it is necessary to understand bellows aging issues in operating nuclear power plants. Age-related degradation of bellows observed in operating nuclear power plants affects the overall performance of the containment system. Preventive maintenance and bellows replacement may be necessary during the current license period to assure that there is no loss of intended function(s) and no unacceptable reduction in safety margins. In addition, continued operation of commercial plants for periods that extend beyond the original 40-year license period may be a desirable option for many US utilities.

US utilities must meet certain Nuclear Regulatory Commission (NRC) requirements before the NRC will extend a facility's operating license. One requirement is to show that the aging of systems, structures, and components (SSCs) within the scope of license renewal is managed so that these items will not degrade to the point where they no longer perform their intended functions. The License Renewal Rule (10 Code of Federal Regulations [CFR] 54, 1996) specifically requires an aging management review (AMR) of containment penetrations.

This research assesses the performance of containment bellows when subjected to Current License Basis (CLB), degraded, and severe accident (SA) conditions. Degradation that results from corrosion, misalignment, and other damage has been considered. The objective of the work is to provide recommendations that can be used for plant in-service inspection, license renewal, and for determining the impact degraded bellows may have on containment capacity.

## 1.2 Background

Bellows are used primarily in steel containments and are of two main types: (1) used in both BWR and PWR containments, process piping bellows vary in size from 15 to 150 cm in diameter. Similar in design, fuel transfer tube bellows typically are about 105 cm in diameter. (2) Vent line bellows, which are used in BWR Mark I containments, range in size from 165 to 315 cm in diameter. Process piping and fuel transfer tube bellows are normally constructed of two plies of SA240, type 304, stainless steel which are separated by a thin wire mesh (~0.25-mm wire diameter). The redundant outer ply was intended to provide a way to check for leakage of the bellows by pressurizing the space between the plies and noting any drop in pressure. In contrast with process piping bellows, the majority of vent line bellows are single ply; approximately 10% are two ply. Single-ply vent line bellows are tested during the integrated leak rate test (ILRT) (10 CFR 50, Appendix J, 1995), because they cannot be tested locally.

The two types of leak rate tests discussed in this report are defined in 10 CFR 50, Appendix J. The ILRT determines the overall leakage rate for the containment though all possible leakage paths, including leakage though containment welds, valves, fittings, and all penetrations of the containment. The ILRT is conducted after the containment has been completed and is ready for operation and periodically thereafter. The other is the local leak rate test (LLRT), which tests for local leak rates across containment penetrations, including containment bellows.

Bellows are designed to accommodate thermal expansion of the pipes and differential movements that occur

- between the containment shell and the concrete shield building in most steel containments,
- between the drywell and wetwell in Mark I BWRs, and
- between the containment building and the transfer tube in Mark IIIs and PWRs.

The design bases for these movements are obtained by summing appropriate combinations of the maximum deformations associated with normal operation, safe shutdown earthquake (SSE), loss of coolant accident (LOCA), settlement, and other loading conditions that could cause differential movement.

The design standards are provided by the Expansion Joint Manufacturers Association (EJMA, 1998). Normally, bellows are conservatively designed to withstand more than 150 cycles of design-basis loadings, with some estimates that 5000 cycles of design-basis loading can safely be withstood. Bellows typically experience only a few cycles from thermal loads at startup and shutdown of the reactor.

In the unlikely event of a SA, the bellows could be subjected to pressure, temperature, and deflections well beyond the design basis. The axial displacement of the bellows would be relatively small until the pressure in the containment vessel caused plastic yielding in the containment wall; plastic yielding will not occur until the containment pressure is well above the design pressure. After the containment begins gross plastic deformation, small increases in pressure would result in large displacements that the bellows must accommodate.

Typically, the bellows will not become fully compressed or elongated until the pressure is several times larger than the design pressure of the containment. In most cases, the bellows would be compressed, but in a few cases, the bellows would be elongated due to the bellows being installed on the inside of the containment. In addition to the bellows being elongated or compressed, they are also deflected transversely due to upward movement of the containment vessel walls caused by the temperature and pressure loadings.

## 1.3 Scope

In performing the tasks described in this report, the following references were reviewed:

- Electric Power Research Institute (EPRI) and other industry-generated reports;
- NRC bulletins, information notices, circulars, generic letters, and reports;
- CFR and Federal Register (FR);
- vendor manuals;
- industry codes and standards (e.g., American Society of Mechanical Engineers [ASME], and EJMA);
- miscellaneous references and technical papers;
- Institute for Nuclear Power Operations (INPO) Nuclear Plant Reliability Data System (NPRDS) and NRC/Oak Ridge National Laboratory (ORNL) License Event Report (LER) databases; and
- input from host utilities for the US Department of Energy (DOE) Containment Penetrations Aging Management Guideline.

Stainless steel expansion joint bellows, used in both BWR and PWR containments, have been evaluated in this study. Applicable codes and standards for the design, testing, and repair of bellows were considered during this review. Locations where bellows typically are used, bellows type, and material are described in this report. This information provides a baseline for the Chapter 6 assessment of risk significance of bellows leakage.

Compilation of instances of bellows cracking, excessive leaking, or structural integrity considerations as well as testing issues and other degradation problems are reviewed and summarized. Data on failure of bellows from INPO's NPRDS, and the NRC/ORNL LER databases and available literature are described. A review of NRC information notices, bulletins, and circulars was also conducted to identify age-related failures. Detailed evaluations of the reduction in pressure capacity caused by the damage were not made at this time; however, conditions observed have been described.

An assessment is made about the procedures that have been used to qualify bellows for continued service after aging degradation has occurred, such as nonlinear finite element analysis, elastic-plastic fracture mechanics, and leak rate projection. Chapter 7 provides recommendations for additional criteria that may need to be considered to ensure the bellows meet their functional and structural design requirements for the remainder of their design life, as well as for additional periods of time if the operating license is extended.

Finally, an assessment has been made about the strengths and weaknesses of the in-service inspection programs used.

# 2. BELLOWS DESCRIPTION

## 2.1 Containment Structure

Containment designs for the 110 nuclear power plants operating in late 1996 are identified in Table 2-1 (BWRs) and Table 2-2 (PWRs). Typically, concrete containments have an interior steel liner plate. Similarly, steel containments are surrounded by a concrete shield wall. Therefore, independent of the containment design, containment penetrations pass through a relatively thin steel wall and a relatively thick concrete wall. In a concrete containments, the concrete wall, liner, and penetration nozzle move together as primary containment temperature and pressure vary. This is not the case for a steel containment where the containment vessel and penetration nozzle move independently of the fixed concrete shield wall.

## 2.2 Containment Bellows

Some piping penetrations for steel containments and some fuel transfer tube penetrations contain metallic expansion joints that are pressure boundary components for the primary containment. In addition, vent line bellows in BWR Mark I containments are also pressure boundary components for the primary containment. These stainless steel expansion joints contain single or double bellows that provide for differential movement at the penetrations. This is necessary to accommodate construction misalignment, potential settlement differentials, and movement caused by temperature, pressure, or other operational or accidental loadings. Most vent line bellows in BWR Mark I containments are single ply, and most process piping and fuel transfer tube bellows in BWR and PWR containments are double ply with four to eight convolutions. The thickness of a single ply is 0.76 to 1.52 mm. The double-ply bellows generally have a tight stainless mesh to separate the plies. Two current issues associated with these stainless steel bellows are

- potential for degradation due to fatigue and transgranular stress corrosion cracking (TGSCC) (Brown and Tice, 1993; Berg and Brown, 1995); and
- inability to leak test the double-ply bellows with air pressure between the plies (Brown and Tice, 1993; NRC, 1992).

Numerous types of bellows are installed in US commercial nuclear power plants. The common bellows manufacturers identified during the preparation of this report are Chicago Bridge and Iron, Pathway, Parker Hannefin, and Tube Turns.

Piping systems that penetrate the primary containment can be designed with the containment penetration as a pipe anchor location (Figure 2-1). The expansion joint shown in the pipe sleeve outside the containment (Figure 2-1) relieves thermal stresses that would develop from differential thermal movement of the pipe and sleeve. In addition, the expansion joint can accommodate some misalignment of the penetration centerline and reduce transverse loadings generated by piping outside containment.

Typical piping penetrations with bellows for steel containments are shown in Figure 2-1 (annulus between steel containment and shield wall) and Figure 2-2 (small gap or compressible material between steel containment and shield wall). In these situations, a significant amount of pipe movement must be accommodated at the penetration to avoid high loads at the connection to the containment. The sleeved opening in the concrete shield wall is relatively unimportant as long as it is large enough not to interfere with the penetration design. The double bellows stainless steel expansion joints in the penetration sleeve can absorb both axial and transverse movement. The guard pipe protects the bellows from damage in the case of a pipe break. The primary containment atmosphere resides between the process pipe and the guard pipe and between the guard pipe and the sleeve. The multiple flued head (part of the piping), the sleeve, and the expansion joint all represent primary containment pressure boundary barriers. Figure 2-3 shows typical piping penetrations with bellows in concrete containments.

A typical fuel transfer tube, as used in a PWR containment, is shown in Figure 2-4. A fuel transfer tube links the primary containment refueling canal with the fuel transfer canal and spent fuel pit in the fuel handling building. The fuel transfer tube is a shielded underwater pathway for moving spent fuel assemblies out of primary containment during refueling operations. Expansion joints are required to accommodate both construction misalignment and differential movement between the refueling canal, containment structure, and spent fuel pit.

## Table 2-1. BWR Primary Containment List

| Unit | Primary Containment Designation* | | | | | | | |
|---|---|---|---|---|---|---|---|---|
| | Mark 0 | Mark I | | Mark II | | | Mark III | |
| | | a | b | a | b | c | a | b |
| Big Rock Point | 1 | | | | | | | |
| Browns Ferry 1, 2, and 3 | | 3 | | | | | | |
| Brunswick 1 and 2 | | | 2 | | | | | |
| Clinton 1 | | | | | | | | 1 |
| Cooper | | 1 | | | | | | |
| Dresden 2 and 3 | | 2 | | | | | | |
| Duane Arnold | | 1 | | | | | | |
| Edwin I. Hatch 1 and 2 | | 2 | | | | | | |
| Fermi 2 | | 1 | | | | | | |
| Grand Gulf 1 | | | | | | | | 1 |
| Hope Creek | | 1 | | | | | | |
| James A. FitzPatrick | | 1 | | | | | | |
| LaSalle 1 and 2 | | | | | | 2 | | |
| Limerick 1 and 2 | | | | | 2 | | | |
| Millstone 1 | | 1 | | | | | | |
| Monticello | | 1 | | | | | | |
| Nine Mile Point 1 | | 1 | | | | | | |
| Nine Mile Point 2 | | | | | 1 | | | |
| Oyster Creek | | 1 | | | | | | |
| Peach Bottom 2 and 3 | | 2 | | | | | | |
| Perry 1 | | | | | | | 1 | |
| Pilgrim | | 1 | | | | | | |
| Quad Cities 1 and 2 | | 2 | | | | | | |
| River Bend 1 | | | | | | | 1 | |
| Susquehanna 1 and 2 | | | | | 2 | | | |
| Vermont Yankee | | 1 | | | | | | |
| WNP-2 | | | | 1 | | | | |
| Total Units (37) | 1 | 22 | 2 | 1 | 5 | 2 | 2 | 2 |

*Notes:
Mark 0: Large, dry, steel sphere.
Mark I a: Steel drywell and wetwell.
Mark I b: Reinforced concrete drywell and steel wetwell.
Mark II a: Steel drywell and wetwell.
Mark II b: Reinforced concrete drywell and wetwell.
Mark II c: Reinforced concrete drywell and post-tensioned concrete wetwell.
Mark III a: Reinforced concrete drywell and steel wetwell.
Mark III b: Reinforced concrete drywell and wetwell.

## Table 2-2. PWR Primary Containment List

| Unit | Primary Containment Designation* | | | | | | | | | | |
|---|---|---|---|---|---|---|---|---|---|---|---|
| | Ice | | Post-tensioned Concrete | | | | Reinforced Concrete | | Steel | Sub-Atmosphere | |
| | a | b | a | b | c | d | a | b | | a | b |
| Arkansas Nuclear One 1 and 2 | | | | | 2 | | | | | | |
| Beaver Valley 1 and 2 | | | | | | | | | | 2 | |
| Braidwood 1 and 2 | | | | | 2 | | | | | | |
| Byron 1 and 2 | | | | | 2 | | | | | | |
| Callaway 1 | | | | | 1 | | | | | | |
| Calvert Cliffs 1 and 2 | | | | | 2 | | | | | | |

**Table 2-2. PWR Primary Containment List (continued)**

| Unit | Primary Containment Designation* | | | | | | | | | | |
|---|---|---|---|---|---|---|---|---|---|---|---|
| | Ice | | Post-tensioned Concrete | | | | Reinforced Concrete | | Steel | Sub-Atmosphere | |
| | a | b | a | b | c | d | a | b | | a | b |
| Catawba 1 and 2 | 2 | | | | | | | | | | |
| Comanche Peak 1 and 2 | | | | | | | 2 | | | | |
| Crystal River 3 | | | | | 1 | | | | | | |
| D.C. Cook 1 and 2 | | 2 | | | | | | | | | |
| Davis-Besse 1 | | | | | | | | | 1 | | |
| Diablo Canyon 1 and 2 | | | | | | | 2 | | | | |
| Fort Calhoun | | | | 1 | | | | | | | |
| Ginna | | | 1 | | | | | | | | |
| H.B. Robinson 2 | | | 1 | | | | | | | | |
| Haddam Neck | | | | | | | 1 | | | | |
| Indian Point 2 and 3 | | | | | | | 2 | | | | |
| Joseph M. Farley 1 and 2 | | | 2 | | | | | | | | |
| Kewaunee | | | | | | | | | 1 | | |
| Maine Yankee | | | | | | | 1 | | | | |
| McGuire 1 and 2 | 2 | | | | | | | | | | |
| Millstone 2 | | | | | | 1 | | | | | |
| Millstone 3 | | | | | | | | | | | 1 |
| North Anna 1 and 2 | | | | | | | | | | 2 | |
| Oconee 1, 2, and 3 | | | | | 3 | | | | | | |
| Palisades | | | | | 1 | | | | | | |
| Palo Verde 1, 2, and 3 | | | | | 3 | | | | | | |
| Point Beach 1 and 2 | | | | | 2 | | | | | | |
| Prairie Island 1 and 2 | | | | | | | | | 2 | | |
| Salem 1 and 2 | | | | | | | 2 | | | | |
| San Onofre 2 and 3 | | | | | 2 | | | | | | |
| Seabrook 1 | | | | | | | | 1 | | | |
| Sequoyah 1 and 2 | 2 | | | | | | | | | | |
| Shearon Harris | | | | | | | 1 | | | | |
| South Texas Project 1 and 2 | | | | | 2 | | | | | | |
| St. Lucie 1 and 2 | | | | | | | | | 2 | | |
| Summer | | | | | 1 | | | | | | |
| Surry 1 and 2 | | | | | | | | | | 2 | |
| Three Mile Island 1 | | | | | 1 | | | | | | |
| Turkey Point 3 and 4 | | | | | 2 | | | | | | |
| Vogtle 1 and 2 | | | | | 2 | | | | | | |
| Waterford 3 | | | | | | | | | 1 | | |
| Watts Bar 1 | 1 | | | | | | | | | | |
| Wolf Creek 1 | | | | | 1 | | | | | | |
| Zion 1 and 2 | | | | | 2 | | | | | | |
| Total Units (73) | 7 | 2 | 4 | 1 | 32 | 1 | 11 | 1 | 7 | 6 | 1 |

*Notes:

1. Ice condenser a: Steel cylinder with concrete shield building.
   Ice condenser b: Reinforced concrete cylinder with steel lining.
2. Post-tensioned concrete a: Concrete cylinder with steel liner, 1-D vertical tendons.
   Post-tensioned concrete b: Concrete cylinder with steel liner, diagonal tendons.
   Post-tensioned concrete c: Concrete cylinder with steel liner, 3-D tendons.
   Post-tensioned concrete d: Concrete cylinder with steel liner, 3-D tendons plus secondary containment.
3. Reinforced concrete a: Concrete cylinder with steel liner.
   Reinforced concrete b: Concrete cylinder with steel liner plus secondary containment.
4. Steel: A steel cylinder containment vessel with a concrete enclosure building.
5. Sub-atmospheric a: Reinforced concrete cylinder with steel liner.
   Sub-atmospheric b: Reinforced concrete cylinder with steel liner plus secondary containment.

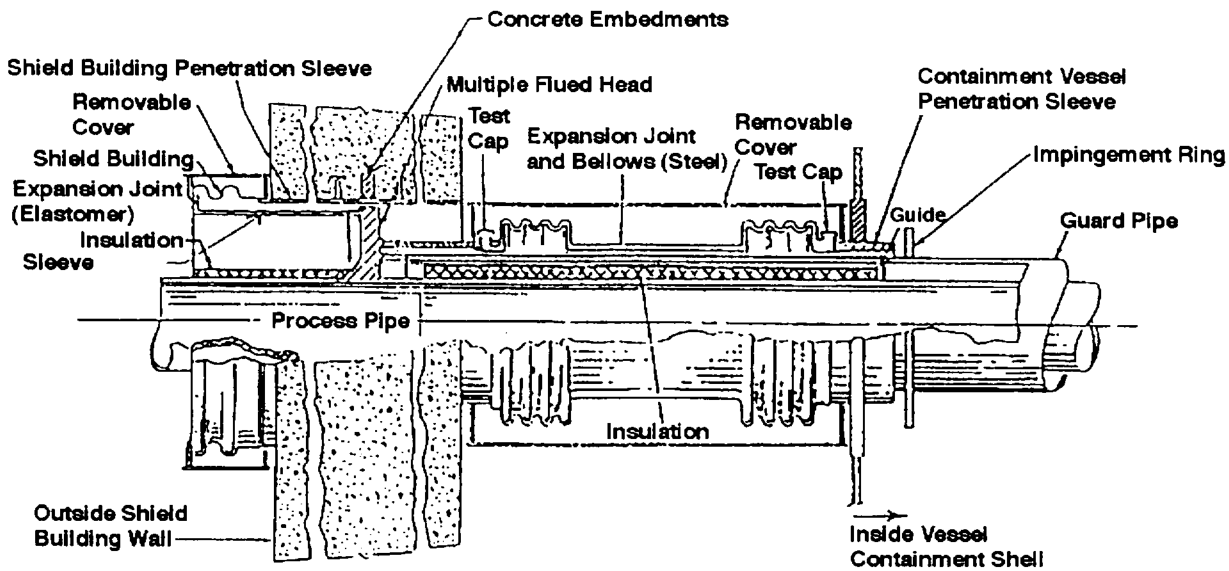


Figure 2-1. Typical Piping Penetration with Bellows.

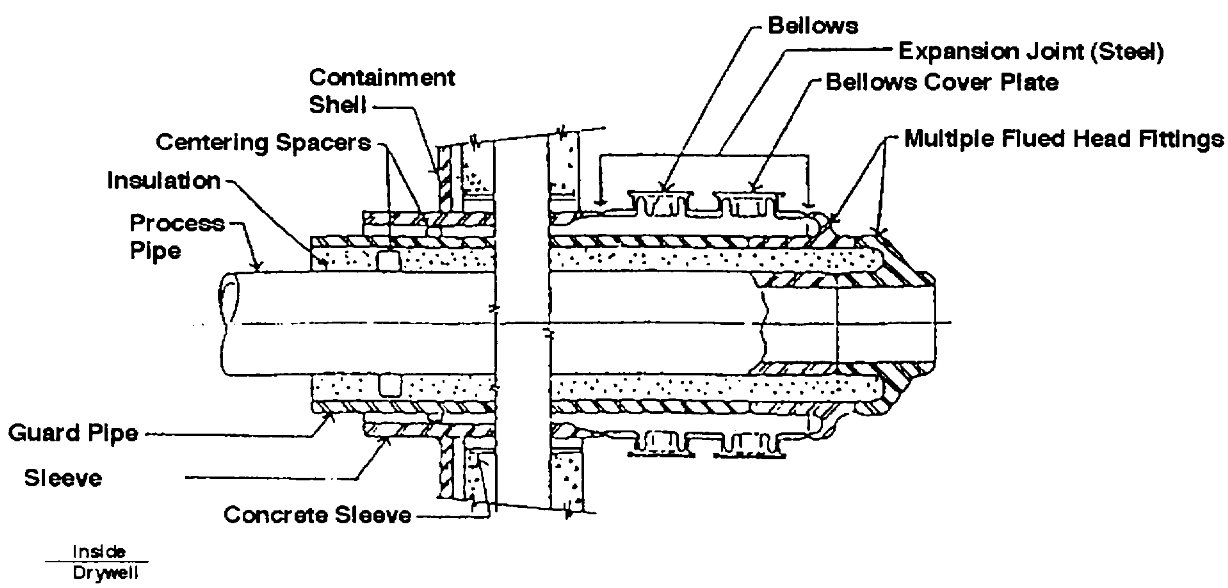


Figure 2-2. Typical Piping Penetration with Bellows - Steel Containment.

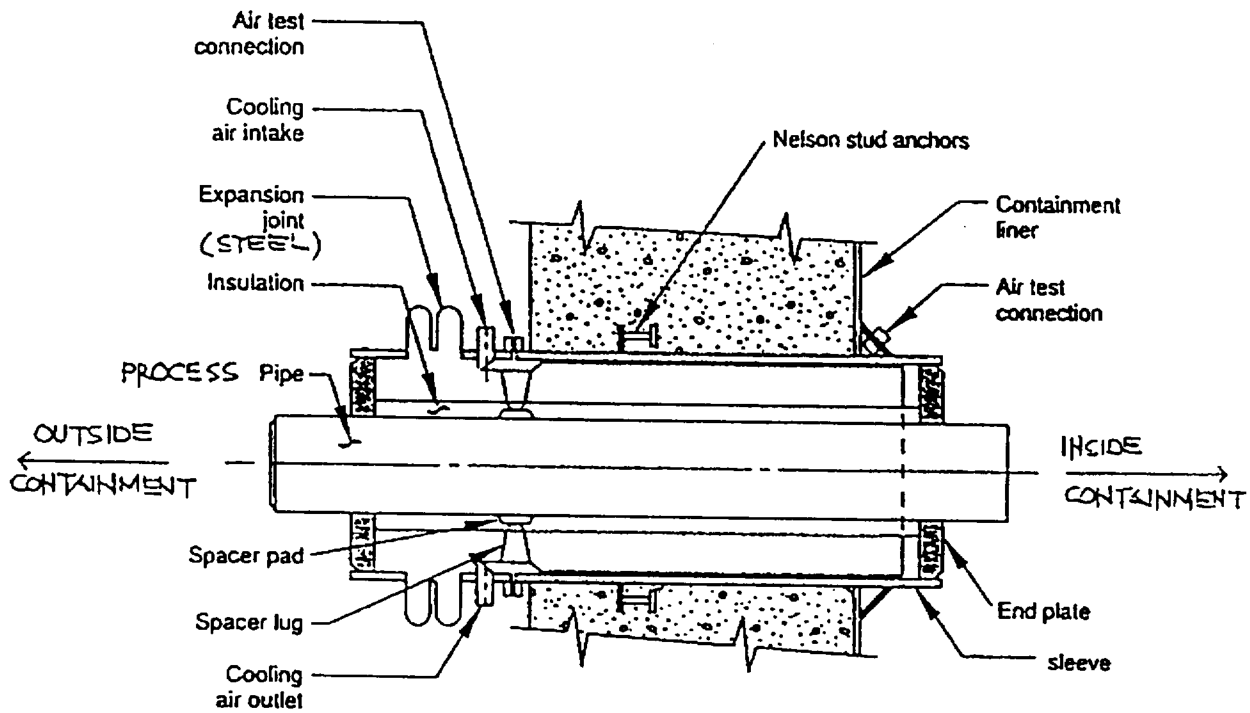


**Figure 2-3. Typical Piping Penetration with Bellows - Concrete Containment.**

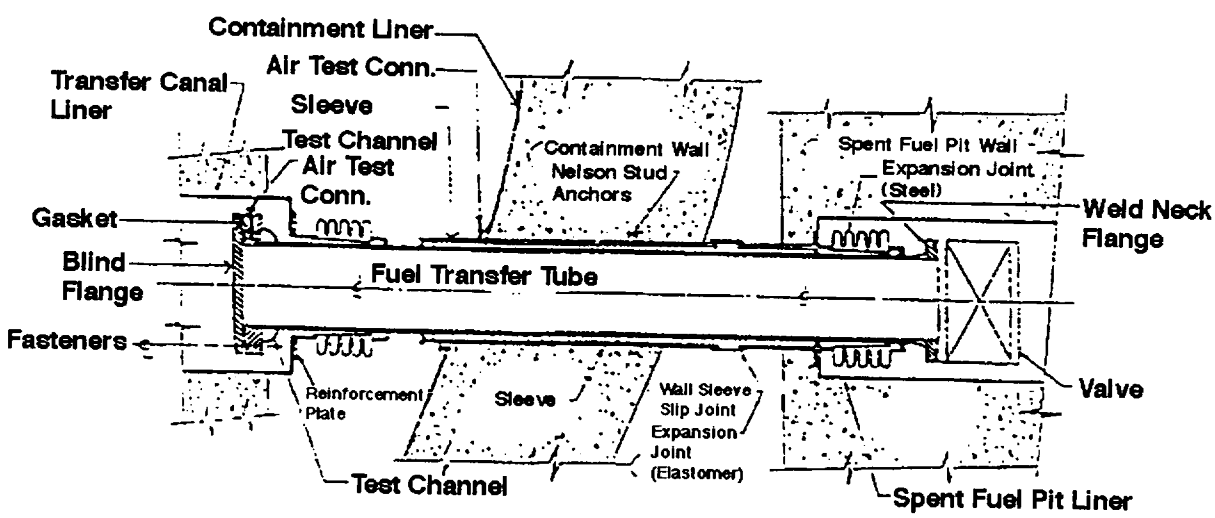


**Figure 2-4. Typical Fuel Transfer Tube – PWR.**

New, unaged stainless steel bellows are tested to characterize and predict their response under design basis accident (DBA) conditions. These studies have shown that the level of extension or compression necessary to cause bellows failure, and resulting loss of containment, is several times greater than that required to satisfy the primary containment design specifications. Analytical studies accompanying these tests, which also provided information regarding the fatigue resistance of these stainless steel bellows, showed that they could withstand many more extension and compression cycles than expected during the operating life of the plant.

Bellows are typically made of austenitic stainless steel SA240, Type 304. However, other designations have been used in a few cases: A-167-63, Type 304 and SB443-alloy 625 Gr 1.

The types of bellows and the numbers of bellows used as part of the primary containment boundary are shown for typical BWR Mark I (Table 2-3), Mark II (Table 2-4), and Mark III (Table 2-5) plants. Similarly, this information is given for a typical PWR Ice Condenser (Table 2-6), Reinforced Concrete (Table 2-7), Steel Containment (Table 2-8), and Sub-Atmospheric plant (Table 2-9).

**Table 2-3. Typical BWR Mark I Bellows (Monticello)**

| Description | Test Type* | Number of Bellows |
|---|---|---|
| Drywell-to-Torus Vent Pipes | A | 8 |
| Main Steam | A, B | 4 |
| Main Steam Drain Line | A, B | 1 |
| Feedwater | A, B | 2 |
| RCIC Steam Supply | A, B | 1 |
| HPCI Steam Supply | A, B | 1 |
| RHR Supply (from Reactor Recirc) | A, B | 1 |
| RHR Return and LPCI | A, B | 2 |
| RWCU Supply | A, B | 1 |
| Core Spray | A, B | 2 |
| RHR Head Spray (to RPV Head) | A, B | 1 |
| Total | | 24 |

*Notes:
1. This list was developed from the Monticello Final Safety Analysis Report (FSAR), Table 5.2-3a, Revision 10 (NRC, 2000b).
2. Type A: 10 CFR 50, Appendix J, ILRT
3. Type B: 10 CFR 50, Appendix J, LLRT

**Table 2-4. Typical BWR Mark II (Nine Mile Point Unit 2)**

| Description | Test Type* | Number of Bellows |
|---|---|---|
| TIP | A, B | 5 |
| Total | | 5 |

*Notes:
1. This list was developed from the Nine Mile Point Unit 2 FSAR (NRC, 2000c).
2. Type A: 10 CFR 50, Appendix J, ILRT
3. Type B: 10 CFR 50, Appendix J, LLRT

**Table 2-5. Typical BWR Mark III Bellows (Grand Gulf)**

| Description | Test Type* | Number of Bellows |
|---|---|---|
| Fuel Transfer Tube | A, B | 1 |
| Total | | 1 |

*Notes:
1. This list was developed from the Grand Gulf FSAR, Table 6.2-49 (NRC, 2000a)
2. Type A: 10 CFR 50, Appendix J, ILRT
3. Type B: 10 CFR 50, Appendix J, LLRT

**Table 2-6. Typical PWR Ice Condenser Bellows (Watts Bar)**

| Description | Test Type* | Number of Bellows |
|---|---|---|
| Fuel Transfer Tube | A, B | 1 |
| Feedwater Bypass | A, B | 4 |
| Feedwater | A, B | 4 |
| Main Steam | A, B | 4 |
| Steam Generator (SG) Blowdown | A, B | 4 |
| Chemical and Volume Control System | A, B | 1 |
| Residual Heat Removal Hot Leg Injection | A, B | 1 |
| RHR Pump Discharge TR B | A, B | 2 |
| Safety Injection System | A, B | 2 |
| Pressure Relief Valve Discharge | A, B | 1 |
| Safety Injection | A, B | 3 |
| RC Drain Tank | A, B | 3 |
| Glycol Supply and Return | A, B | 2 |
| RHR Supply to Tanks | A, B | 1 |
| Maintenance Port | A, B | 2 |
| Total | | 35 |

*Notes:
1. This list was developed from the Watts Bar FSAR, Table 6.2.4-1, Amendment 85 (NRC, 2000f).
2. Type A: 10 CFR 50, Appendix J, ILRT
3. Type B: 10 CFR 50, Appendix J, LLRT

**Table 2-7. Typical PWR Reinforced Concrete Containment Bellows (Palisades)**

| Description | Test Type | Number of Bellows |
|---|---|---|
| No bellows | none | 0 |
| Total | | 0 |

**Table 2-8. PWR Steel Containment Bellows (Prairie Island)**

| Description | Test Type* | Number of Bellows |
|---|---|---|
| Main Steam Header | A, B | 2 |
| Main Feedwater Headers | A, B | 2 |
| SG Blowdown | A, B | 2 |
| RHR Loop Out and In | A, B | 2 |
| CVCS Letdown Line | A, B | 1 |
| Total | | 9 |

*Notes:
1. This list was developed from the Prairie Island FSAR, Table 5.2-1, [(Part A), Unit 1], Revision 13 (NRC, 2000d).
2. Type A: 10 CFR 50, Appendix J, ILRT
3. Type B: 10 CFR 50, Appendix J, LLRT

**Table 2-9. Typical PWR Sub-Atmospheric Bellows (Surry)**

| Description | Test Type* | Number of Bellows |
|---|---|---|
| Fuel Transfer Tube | A, B | 1 |
| Total | | 1 |

*Notes:
1. This list was developed from the Surry FSAR, Table 6.2-49 (NRC, 2000e).
2. Type A: 10 CFR 50, Appendix J, ILRT
3. Type B: 10 CFR 50, Appendix J, LLRT

# 3. OPERATIONAL EXPERIENCE

All bellows used in penetrations are safety related because they are required to prevent or mitigate the consequences of accidents that might result in offsite exposures covered by regulatory guidelines (10 CFR 100, 1962). As noted in Section III.f (page 22477) in the supplementary information portion of the amended license renewal rule (10 CFR 54, 1996), all containment penetrations (i.e., bellows) must be included in the scope of license renewal as "passive" components.

PWR and BWR Mark IIIs include access penetrations termed fuel transfer tubes, which utilize a bellows to allow relative movement between the containment and the transfer tube. Most steel containments, and a few concrete containments, provide metallic bellows to accommodate piping thermal expansion and/or the differential movement between the steel containment shell and concrete shield wall. The BWR Mark I containments have vent line bellows between the drywell and the wetwell.

## 3.1 Operating and Service History

Four information sources were reviewed to gain an understanding of operating and service history. Only documents that pertain to aging of bellows were considered in this report:

(1) A review of NRC generic communications (bulletins, circulars, generic letters, and information notices) was conducted to assess the regulatory perspective of US operating experience. (2) Industry studies and literature were searched for documents relating to penetration operating history and failures. Containment penetration data derived from (3) INPO's NPRDS and (4) NRC's LER databases were also reviewed. Component failures described in these databases were analyzed to identify significant failure mechanisms and their relative likelihood of occurrence.

### 3.1.1 Operating Experience from Nuclear Regulatory Commission Generic Communications

NRC Information Notice 92-20 (1992) documented the inadequacy of performing local leak rate testing for piping penetrations by pressurizing between the plies in two-ply expansion joint bellows. Tests being performed in this manner were Type B LLRTs in accordance with 10 CFR 50, Appendix J (1995). Tests at Quad Cities, Dresden, Perry, and Clinton indicated that these between-the-plies tests could underestimate the leakage across the bellows by a factor of more than 20. The conclusion was that these two-ply bellows were manufactured using a method that inhibited the free flow of air between the bellows plies. The NRC provided exemptions to plants with this type of two-ply bellows. The exemptions specify that an alternative means for testing these bellows is required and test frequencies must be increased until the existing bellows are replaced with testable bellows. Alternative testing includes testing bellows where any level of leakage has been detected using helium leak detection equipment.

### 3.1.2 Operating Experience from Industry Publications

Sams and Trojovsky (1980) analyzed LERs of various containment penetration events submitted to the NRC between January 1, 1972, and December 31, 1978. Penetration components were defined as the penetration assembly and all hardware associated with the normal operation of the penetration. Manufacturer, failure mode, failure mechanism, type, and date of event were recorded.

Sams and Trojovsky (1980) identified 431 potentially applicable LERs, for all penetrations. Of these 431 reports, three contained instances of bellows failures. On December 13, 1974, Dresden Unit 2 reported that a penetration bellows showed excess leakage. The cause was identified as a cracked bellows with numerous pinholes. On June 16, 1975, Zion Unit 2 reported that a penetration bellows was leaking in excess of allowable limits. On August 1, 1977, Dresden Unit 1 failed its ILRT: two bellows had cracks; five bellows had pinhole leaks. Nine failed bellows were reported by Sams and Trojovsky (1980).

Shackelford et al. (1985) presented design data and detailed drawings of piping and access penetrations for 26 nuclear units. The primary reason for gathering the data was to further develop methodologies for analyzing and testing penetration responses under SA conditions. Relative stiffness ratios for large access penetrations were developed. These ratios were used to select those penetration models for which detailed analysis and testing should be performed.

Kulak et al. (1985) discussed analyses of large hot piping penetrations tested to pressures and temperatures beyond their design specification. The analysis also evaluated the fatigue resistance of piping penetration expansion joints. Based on conservative elastic calculations, the stainless steel bellows in these hot piping penetrations should be able to withstand more than 150 major compression cycles.

Greimann et al. (1991) discuss the nondestructive examination of full-penetration welds in the guard pipe and sleeve of hot piping penetrations. Volumetric examination of these welds, using ultrasonic techniques previously approved by the NRC and ASME, to identify flaws found the techniques to be inadequate for flaw detection. Subsequent radiographic examination of these welds identified flaws that exceeded allowable levels. These welds were made from one side with backing rings. It is apparent that this weld configuration did not allow appropriate interpretation of ultrasonic examination results. Licensees were requested to ensure that similar welds in hot piping penetrations are nondestructively examined using radiographic techniques.

Lambert and Parks (1994) performed a series of 13 tests on piping and vent line expansion bellows in a "like-new" condition to various levels and combinations of internal pressure, temperature, axial compression or elongation and lateral deformation. Some of the tests were conducted at elevated temperature (218°C). The test objective was to determine the behavior of the bellows in a SA, with conditions that were well beyond design conditions. Test results showed that bellows in the "like-new" condition are capable of remaining leak tight under extreme combinations of displacement, pressure, and elevated temperatures in both compression and extension.

Lambert and Parks (1995) also tested six piping and vent line bellows that had been artificially aged to produce TGSCC in the bellows. The test objective was to determine the behavior of the corroded bellows in a SA. The bellows were subjected to elevated pressure, temperature, and lateral and axial deformation conditions similar to the "like-new" bellows (Lambert and Parks, 1994). The goal of testing the corroded bellows was to produce corroded conditions evident via visual inspection, but not severe enough to cause the bellows to leak. Lambert and Parks (1994) concluded, "Because of the limited number of corroded specimens and the problems experienced in producing and quantifying the precise amount of desired corrosion, it is impossible to draw any comprehensive conclusions regarding the leak-tight capabilities of bellows with varying degrees of corrosion." However, as one might expect, tests showed that corroded bellows did not perform as well as "like-new" bellows.

### 3.1.3 Operating Experience from Nuclear Plant Reliability Data System Data

Bellows failure data were reviewed. One source of failure data is the INPO's NPRDS database, which can be queried to obtain information such as the type of component, date of discovery, cause category, and a brief description of the event. NPRDS data are not focused directly on component aging; NPRDS does not necessarily address root cause or mechanism of component degradation. Additionally, not all degradations observed during maintenance activities are identified in the database. Not all plants participate in the NPRDS database; those nuclear power plant personnel who have reported may not have provided data for their plant's entire period of operation. As a result of these limitations, the database cannot be used to provide probabilistic information about the reliability of a bellows with respect to age-related degradation. However, the data can be used to corroborate bellows degradation information identified by other sources.

The NPRDS database was searched using component category (i.e., containment penetrations) and subcategory (process piping). Very little degradation and failure data were available in NPRDS for piping penetrations. Thirty-five piping penetration incidents were identified; however, only 12 of the incident reports attributed incidents to leaking stainless steel expansion joint bellows.

In most cases, degradation was identified through leakage tests (10 CFR 50, 1995). Maintenance action, specifically bellows replacement, was required to resolve these degradation incidents.

There were three incident reports that involved fuel transfer tube aging, where the principal degraded subcomponent was a gasket. The root cause was not identified, but the failures were not the metallic bellows.

### 3.1.4 Operating Experience from License Event Report Data

NRC LERs are another source of penetration failure and degradation data. LERs are issued by nuclear plant operators when component failures and plant operating events meet the reporting requirements specified in 10 CFR 50.73 (1995). As with NPRDS data, LERs do not directly record data related to component aging. In addition, the criteria for issuance of an LER do not encompass all component failures. Hence, LER data provide only a partial picture of failure information and the data may or may not be representative of general component failure behavior. LER data can be used to support findings derived from other data sources (e.g., NPRDS, industry studies) and to corroborate postulated aging mechanisms.

Sams and Trojovsky (1980) describe a search of LERs for penetration failures from 1972 through 1978 (see Section 3.1.2). A search of the LER database maintained by ORNL identified 447 abstracts for the period 1980 through 1998 that might potentially apply to primary containment penetrations. Each report was reviewed; in cases where applicability of a given report could not be reliably determined, the report was discarded. Five LERs were applicable to bellows failures in primary containment penetrations. The failures were identified during leak rate testing and surveillance. The causes of the failures were identified as TGSCC. A brief summary follows of the bellows failure information in the LERs for the 1980 through 1998 period. (Section 3.1.2 contains a brief summary of bellows failures in the LERs for the 1972 to 1978 time period.)

<u>Dresden Nuclear Power Station Unit 2</u>

On April 2, 1991, while Unit 2 was shutdown for a short maintenance outage, six drywell bellows penetrations were inspected for cracks. (During the previous Unit 2 refuel outage, eight of the 24 bellows had a positive leakage rate. The inspection lot of six was chosen from these eight based on accessibility and dose.) Through-wall cracks were found on the exterior bellows ply at penetration X-144.

The initial indications on bellows X-144 were identified through a penetrant test (PT). Following the PT examination, the bellows were pressurized and coated with a soap bubble solution to check for through leakage. This inspection revealed additional leak indications undetectable by PT. The bellows were also inspected with helium leak detection equipment. The results of this inspection revealed leakage past both the inboard and outboard bellows plies. The root cause of the indications found on bellows X-144 was attributed to TGSCC. The largest crack was 7/16 inch long.

On January 21, 1993 with Unit 2 in a refuel outage, bellows penetrations X-113, X-125, X-149A, and X-149B were replaced with a new design which provided increased space between the plies. No reason was given for replacing the bellows. This change allowed the total surface of the bellows to be challenged during Type B Local Leak Rate Testing. During the 1993 refuel outage, bellows penetration X-144 was sealed from within the drywell. This planned modification eliminated bellows X-144 as an Appendix J Type B testable volume (10 CFR 50, 1995).

<u>Dresden Nuclear Power Station Unit 3</u>

During the September 1991 refueling outage, 91 penetrations were tested, including valves, electrical penetrations, bellows, and other primary containment penetrations. Two bellows penetrations, X-105A and X-107B, required repair and were replaced with a new design which provided increased space between the plies. These replacements allow the total surface of the bellows to be challenged during Type B Local Leak Rate Testing.

<u>Quad Cities Unit 1</u>

In the past, double-ply bellows have been Type B LLRT tested by pressurizing between the inner and outer plies of the bellows using a local test line. The X-25 bellows were tested using this method on November 14, 1990, and the leakage rate was 4.3 SCFH. Another test was conducted on February 26, 1991, by pressurizing between the plies; the leakage rate was 6.0 SCFH. The bellows were visually inspected for leakage, and several tiny pinhole cracks were discovered in the outer ply by using soap bubble solution.

During the pressurization phase of the Type A test (ILRT) on February 28, 1991, larger visible cracks were discovered in the outer layer of the X-25 bellows. Following the ILRT, a LLRT was performed, and the leakage rate of the bellows was 6.0 SCFH. Because the leakage of the bellows during the ILRT was believed to be much higher than that measured during the Type B LLRT test, a plate was welded across the purge exhaust pipe inside the drywell to allow testing the bellows under the same conditions as the ILRT. The bellows were tested from inside the containment on March 12, 1991, by pressurizing through the purge exhaust pipe to the inner ply of the bellows; measured leakage was 137.4 SCFH. The X-25 bellows was then cut out and replaced with a new bellows.

<u>H.B. Robinson Steam Electric Plant Unit 2</u>

During the 1984 steam generator (SG) replacement outage, the three SG blowdown lines (penetration numbers S-24, S-26, and S-30) were replaced with larger diameter blowdown lines to accommodate a higher blowdown rate for the new SGs, and the insulation was replaced with insulation sized to fit the larger diameter pipe. In 1986, a penetration bellows failure occurred in SG blowdown line penetration S-24. The bellows was subsequently replaced during 1987; following an analysis of the defective bellows, the failure mechanism was determined to be TGSCC. At that time, recommendations were made to consider future replacement with materials resistant to TGSCC. During 1989, leaks were again detected in the penetrations bellows for all three penetrations, and repairs were scheduled for 1990. During these repairs, an undetermined quantity of water was discovered inside the S-30 penetration, and a chloride-free insulation was installed on the piping in penetration S-30. However, TGSCC-resistant materials were not considered for the other two penetrations (S-24 and S-26).

On July 21, 1995, inspection of the penetration S-24 bellows revealed a crack approximately an inch or two in length; further analysis revealed the presence of chlorides inside the penetration. It was concluded that condensation of water from the PPS-supplied air inside the penetration wetted the pipe insulation, and transported the chlorides contained in the insulation material to the penetration bellows. The presence of these chlorides on the stainless steel material of the penetration bellows caused the bellows to fail due to TGSCC. Penetration S-24 bellows assembly was replaced by July 31, 1995. Containment penetrations S-24, S-26, and S-30 were re-evaluated and re-classified from 10 CFR 50, Appendix J, "Type B" penetrations (i.e., tested to detect and measure local leakage) to 10 CFR 50, Appendix J, "Type A" penetrations (i.e., tested to measure the overall integrated leakage of the primary containment).

### 3.1.5 Nuclear Plant Reliability Data System and License Event Report Comparison

NPRDS and LER data compare well for the bellows. Both databases show a small number of failures (12 from NPRDS and 16 from LERs), with similar aging mechanisms (e.g., TGSCC). Twelve bellows failures were reported in the NPRDS during the 1974-1992 time period, and sixteen bellows failures were reported in the LERs (9 bellows failures were reported by Sams and Trojovsky (1980) for the 1972–1978 period, and 7 bellows failures were found during a search for the 1980–1998 period). From reading the LER reports, it is obvious that bellows have been replaced previously that are not included in the total count of 16 bellows failures discussed here.

## 3.2 Operational Experience Summary

Four primary sources of information (Section 3.1) were used in this report to characterize penetration historical performance. Each provides a consistent perspective on bellows aging. No statistical inferences can readily be drawn concerning bellows failure probability or rate, because not all cases of bellows degradation have been identified. Despite this limitation, several general observations can be stated:

- A number of bellow failures have occurred throughout the industry, with these failures primarily caused by TGSCC.
- Degradation of bellows usually has been identified through leak rate tests.
- After leak tests have identified penetrations with excessive leak rates, visual methods have identified pin-holes and cracks (with the aid of PT in some bellows). Visual methods did not find TGSCC damage to some bellows in penetrations with leakage that was within allowable limits, but larger than expected. (It was not clear from the literature whether the leakage was through the bellows, or another part of the penetration component, or whether the cause of leakage was even determined.)
- After leak tests have identified penetrations with excessive leak rates, helium leak detection equipment has been utilized successfully to locate leak locations.
- Given an aggressive environment (i.e., a humid environment and nearby insulation with chlorides, so that chlorides can leach out of the insulation and onto the bellows), bellows failure can occur over a period of a few years.
- Bellows failures have occurred in nuclear power plants that are fairly new, as well as in older plants. Additionally, new bellows used to replace older bellows have also developed TGSCC and have had to be replaced themselves. However, many older bellows have not experienced TGSCC. Therefore, it appears that the environment (i.e., presence of chlorides or other chemical agents) is a major factor, and bellows age, by itself, is not a good indicator of the likelihood of TGSCC damage.

# 4. AGING MECHANISMS AND EFFECTS ON BELLOWS

Bellows aging mechanisms have been identified, with consideration given to the design and operating history of bellows, along with relevant industry research, information, and reports. For purposes of this document, the significance of an aging mechanism centers around whether the aging mechanism results in loss of bellows functionality during the current or license renewal period.

## 4.1 Electrochemical Aging Mechanisms and Effects

This report focuses specifically on bellows and the stainless steel used in their construction. Low-carbon steels are often used in the penetration component, and stainless steel sometimes is welded to the low-carbon steel. In carbon steel/stainless steel combinations that have surfaces inadvertently wetted or wetted under normal operation (i.e., in the fuel transfer tube), the carbon steel surface could be subjected to galvanic corrosion that can cause significant amounts of material loss over small areas. Therefore, corrosion of any carbon steel material that is welded to stainless steel in the penetration assembly is potentially significant. However, evaluation of possible damage to non-bellows parts used in the penetration assembly is beyond this project's scope.

The primary aging mechanism that is of concern for bellows is TGSCC—an aging mechanism that is associated primarily with austenitic stainless steel materials. All bellows are made from austenitic stainless steel. The three conditions necessary for TGSCC are a sensitized material (resulting from the manufacturing and/or installation process), a high tensile stress (residual or applied), and a suitably aggressive environment (moist or wetted environment, or in an environment contaminated with small amounts of chloride, fluoride, or sulfates, or high temperature). Higher temperature encourages SCC, which generally will not occur below 121°C (250°F) unless aggressive contaminants, such as chloride, are present.

The expansion joints are subjected to high tensile stresses, particularly in the convolutions of the bellows, where high local residual stresses are present at all times. These expansion joints may be subject to water environments part of the time, as well as be contaminated with materials (fluids, chlorides, sulfides, etc.) that encourage initiation of TGSCC and crack growth. Several instances of TGSCC of piping penetration expansion joint bellows have occurred at nuclear plants in the last 20 years. Therefore, TGSCC for the piping, vent line bellows, and fuel transfer tube stainless steel expansion joints can be significant.

The presence of contaminants in the atmosphere or deposited on component surfaces can lead to accelerated aging. These contaminants include chlorides, fluorides, peroxides, and sulfur compounds. There are several potential sources of these contaminants at the bellows:

- inadequate cleaning after completion of the initial containment penetration installation.
- in-service inspection, testing, and maintenance activities without adequate care or cleanup.
- cleaning compounds used in other areas of the containment inadvertently being spread into containment penetration components.
- residual material, from welding processes (smoke and other particulates) associated with penetration components during initial installation or in-service maintenance, remain on or within penetration components. The specific welding activity of concern is associated with the installation and repair of stainless steel expansion joints for mechanical piping penetrations.
- decomposition of nonmetallic components such as thermal insulation or surface coatings. For example, chloride ions present in some types of lagging may be leached from the insulation during exposure to moisture or wetting.

## 4.2 Microbiologically Influenced Corrosion

Although Microbiologically Influenced Corrosion (MIC) has been found in stainless steel piping, the bellows environment is not conducive to this type of damage. No report of MIC damage to bellows was found during the search of the LER and other databases.

## 4.3 Stresses in Bellows

Stresses in the bellows, caused by normal operation and environmental conditions, have a direct impact on the existence and progression of aging-related degradation of the bellows. Thus, it is important to understand the behavior of bellows materials when subject to various stresses in order to design and operate the bellows component satisfactorily, and to develop methods for detecting and mitigating component degradation.

Stresses in the bellows result from static loads, cyclic loads, differential movement (settlement, thermal expansion/contraction, or misalignment during construction) and residual stresses from forming the bellows. Physical damage (during installation, maintenance, or operation) can cause small dents or scratches that lead to local stress concentrations. Although not directly producing stress, environmental conditions such as a chloride contaminant on the surface of the bellows can exacerbate the effects of the stresses and result in more rapid deterioration. For example, chloride contaminants and residual stresses may combine to cause TGSCC.

In Section 3, aging mechanisms were substantiated where possible through evaluation of operating history. However, operating experience in benign environments is not totally conclusive. Accident environments may result in additional stresses. For example, Lambert and Parks (1995) tested bellows that were degraded in the laboratory which were leak tight when subjected to internal pressure with no axial or lateral displacements. However, when the degraded bellows were subjected to large axial and lateral displacements, cracks initiated and grew in the bellows, and the bellows leaked.

### 4.3.1 Static Loads

Bellows are designed to withstand specific types of static loads. These include

- pressure loads from pressurization of the primary containment,
- forces from thermal movement of piping during system heatup and cooldown (only for hot piping penetrations),
- forces from differential thermal movement of structures through which the penetration extends,
- forces from supporting the weight of penetration components or other adjacent components, and
- residual stresses in the bellows convolutions caused by the bellows forming process.

Provided that these static loads are within those specified for design of the penetration, they should not cause component degradation. However, there are two situations of further concern (detailed in Section 4.3.3): (1) in-service differential settlement of the multiple structures that are in contact with the penetration joint, and (2) axial misalignment of the entire penetration or the penetration expansion during initial installation or subsequent maintenance activity.

The most significant static load is the residual stresses in the bellows. These residual stresses, which are at the yield stress level in many locations, are a result of cold forming the bellows. Because the bellows are not annealed after they are formed, the residual stresses remain for the bellows' entire lifetime. These large residual stresses can combine with environmental conditions, such as chloride contamination on the surface of the bellows, and cause TGSCC.

Several of the static loads described above can be applied periodically and are, therefore, also cyclic loads (with the exception of the forces from weight). These cyclic loads can affect many components in all types of penetrations.

### 4.3.2 Cyclic Loads

Stainless steel expansion bellows have been designed to withstand cyclic loads, and therefore, should not be subject to fatigue cracks, unless the number and/or magnitude of the loads exceeds those specified for design. Generally, the design basis loads (i.e., thermal loads from startup and shutdown) cause small cyclic plastic strains, but the plastic strains and expected number of cycles are well below the number required to cause bellows degradation or failure. However, this may not be the case if the bellows has been degraded by TGSCC or subjected to mechanical abuse resulting in scratches in the bellows. When these types of degradation exist, fatigue cracking and failure may be accelerated by cyclic loads.

Fatigue of metallic expansion bellows is primarily associated with the cyclic load stress described above. The cyclic loadings are from periodic application of pressure loads and forces due to thermal movement of piping transmitted through penetrations and structures to which the penetrations are connected. The effects of aging (crack initiation and growth) are associated with the number and magnitude of the cyclic loads and the stress levels resulting from these loads. In general, the fatigue cracks will become larger on the bellows surface where large residual strains have already caused TGSCC to occur.

Pressure testing (ILRT) of the entire containment and LLRTs for individual penetrations require periodic application of pressure loads. Periodic piping thermal loads are applied to hot piping penetrations when the piping systems are heated up and cooled down and when significant changes in fluid temperature occur within the piping during operation. These periodic structure thermal loads are generally associated with steel containments where there is differential movement between the steel shell and the biological shield wall during heatup and cooldown of the primary containment environment, which generally coincides with power plant heatup and cooldown.

Fatigue failures of expansion joint bellows have occurred at nuclear power plants. However, the root cause is not fatigue but damage by TGSCC that weakens the steel and allows fatigue to accelerate the crack growth.

### 4.3.3 Differential Movement

Differential movement that must be accommodated by the bellows can be caused by

- settlement of the structures to which the mechanical penetrations are attached,
- misalignment during construction, and
- thermal expansion and contraction of the structures the bellows connect.

Because the bellows are designed to accommodate large relative displacements, small amounts of permanent differential movement caused by settlement or misalignment will not cause a significant reduction in capability, unless TGSCC damage has already significantly degraded the bellows. Bellows can easily accommodate a permanent differential movement of some amount that is small in comparison to the total capability.

For example, a permanent differential movement of 0.5 in. would be small compared to an overall capability of 10 in. In this example, the primary effect would be that only 9.5 in. of additional movement can be accommodated. Because the travel that bellows are capable of accommodating is far greater that the design limits, a small permanent differential movement from construction misalignment or differential settlement will not negatively impact safety for DBAs unless other damage, such as TGSCC, has also accumulated. However, in the event of a SA that is far larger than the design level, there may be some small reduction in overall capacity. In the example above, if a bellows needed to accommodate 10 in. of displacement during a SA but was able to accommodate only 9.5 in., a small reduction in bellows capacity would result.

A small permanent differential movement that the bellows must accommodate does not increase the stresses in the bellows. The convolutions of the bellows already have residual stresses that are at the yield stress level. Although small movements of the bellows cause additional yielding, the stress values do not increase. If damage to the bellows has occurred from another means, such as a scratch or TGSCC, then small movements could cause a crack to initiate. However, small differential movement would be a problem only if other significant damage such as TGSCC had already accumulated. In which case, bellows functionality had been compromised before the differential movement occurred.

Misalignment adjustments are made when the bellows are installed new, and no TGSCC exists. Therefore, the issue of permanent differential movement from settlement or misalignment during construction will usually not be a concern for stainless steel expansion bellows. However, any relative movement that is accommodated by the expansion bellows should be evaluated to ensure that the measured movement is small compared to the total movement the bellows could accommodate. The motion in the axial direction of the bellows, as well as lateral to the axis of the bellows, should be evaluated.

Thermal expansion and contraction of the structures the bellows connect will also cause differential movement. This movement is quite small and will not cause a premature failure unless other damage to the bellows, such as TGSCC, has already occurred. This type of loading, which occurs during startup and shutdown cycles, is detailed in Section 4.3.2.

## 4.4 Degradation Related to Installation, Maintenance, or Operation

Degradation during penetration assembly or component installation is not in itself an aging mechanism, but it can adversely affect the longevity of the penetration. Installation degradation normally is controlled or prevented through use of approved procedures, controls, and standards during installation or maintenance. Equipment damage usually occurs as a result of failure to use or properly implement procedures, or, in a limited number of cases, from improper equipment design. Examples of such degradation include accidentally scratching or denting expansion joint surfaces.

Incidental mechanical degradation can, by itself, cause a functional failure of a component. Mechanical damage is considered because it could occur for some types of penetrations and may not be detected and reported. This damage might then lead to accelerated aging.

## 4.5 Radiation Effects

During normal operating conditions in containment areas, the normal radiation dose rates (all sources) may range from about 0.1 to 1000 Gy/hr (0.01 to 100 rad/hr), depending on plant type (EPRI, 1986). Although some bellows are located in high-dose spaces, the stainless steel used in their construction has a high damage threshold. Metals generally have damage thresholds on the order of $1 \times 10^{19}$n/cm$^2$ (approximately $2.5 \times 10^8$ Gy or $2.5 \times 10^{10}$ rad) (NASA, 1969). A conservative threshold dose of $1 \times 10^{17}$ n/cm$^2$, or 3 MGy (300 Mrad) gamma, can be set for metals used in penetration components. Below this dose, few or no aging effects should occur in such materials. Few locations within primary containment would experience this level of exposure. Therefore, radiation effects are not a concern for the metallic bellows.

# 5. INSPECTION AND EVALUATION OF BELLOWS

## 5.1 Requirements for Continued Operation during Current License Period

Maintenance, surveillance, and condition monitoring are performed to ensure that the characteristics or attributes of components that are essential for operation are maintained. The activities performed during the operation, maintenance, testing, and condition monitoring of piping penetrations may be generally classified as

- visual or physical inspection,
- leak rate testing,
- cleaning, and
- component repair or replacement.

In addition to individual maintenance, surveillance, and condition monitoring techniques, programs resulting in the management of aging effects on containment penetration components were considered. For example, maintaining compliance with 10 CFR 50.49 (1983) can be considered an aging management program for qualified containment penetrations because the net effect is to maintain adequate post-accident component functionality through establishment of a qualified-life standards for components. All nuclear power plants have existing environmental-qualification programs that include some portion of the total penetration population within their scope. Similarly, a structured bellows LLRT program based on plant-specific operating experience, and coupled with root cause analysis and appropriate corrective action, could be considered an aging management program. Note that some existing plant activities not specifically focused on containment penetration aging management, may be credited with addressing aging effects:

Preventive Maintenance Programs

Although periodic inspection of expansion joint bellows is performed under the surveillance programs (Section 5.1.1), some periodic cleaning and refurbishment or replacement of these bellows may be performed under the preventive maintenance program.

Maintenance Rule

Effective July 1996, nuclear power plant owners have implemented the Maintenance Rule (10 CFR 50.65, 1996). Because of their safety-related function, containment penetrations are included within the scope of the Maintenance Rule—a performance-oriented rule that establishes reliability goals for both systems and individual components. The investigation of maintenance preventable functional failures (MPFFs) will support aging management for stainless steel bellows. An MPFF for a piping penetration may be a crack in the bellows that results in it being unable to meet the leakage acceptance criteria during a LLRT. The program requires that MPFFs be monitored, that the root cause of the failure be investigated, and that a corrective action plan be developed and implemented.

Table 5-1 summarizes programs used to detect damage to bellows.

### 5.1.1 Surveillance Programs

Surveillance programs provide for visual inspection of primary containment accessible surfaces, including the mechanical penetrations, once each operating cycle to satisfy Technical Specification requirements. Bellows are inspected for discoloration, scratches, cracking, leaking, and any other abnormal appearance. Additionally, the presence of dirt, dust, contamination, moisture/humidity, chemicals, or corrosion products may indicate degradation. Inspection results are recorded and corrective action is taken as necessary.

Many of the nuclear power plants commissioned after 1980 have primary containment vessels and mechanical penetrations designed in accordance to Section III, Class MC and Class CC of the ASME Boiler and Pressure Vessel (B&PV) Code (ASME, 1992a). In 1986, the ASME amended Section XI of the B&PV Code, adding Subsection IWE, to include in-service inspection requirements for containment vessels, including mechanical penetrations. Although Subsection IWE is a formalization of the Surveillance Programs already being performed by operating nuclear power plants, the inspection requirements are significantly more rigorous. In 10 CFR 50.55a(b) (2)(ix – x) (1997), the NRC formally required licensees to use the 1992 edition with the 1992 Addenda of Subsection IWE in their nuclear power plant ISI programs. The requirements of Subsection IWE do not have to be implemented until September 9, 2001.

Although periodic visual inspections provide useful observations for many penetrations, they have not been

**Table 5-1. Aging Management Programs for Bellows**

| No. | Aging Management Activity | When Activity is Planned/Frequency of Activity |
|---|---|---|
| | | **Common Programs** |
| 1 | Surveillance Program | Once each operating cycle |
| 2 | 10 CFR 50, Appendix J leak rate testing | ILRT: three times in a 10-year period<br><br>LLRT: every refueling outage |
| 3 | ASME Section XI In-service Inspection Program, Article IWE | Refer to ASME B&PV Code, Section XI, Article IWE-2000, "Examinations and Inspection" (ASME, 1992). |
| 4 | Preventive Maintenance Program | Frequency based on written plant procedures and depends on component. |
| 5 | Periodic plant walkdowns | Performed daily and prior to plant restart after refueling, etc. |
| | | **Less Common Programs** |
| 6 | Settlement Trending Program | Settlement monitoring based on accepted methods (EPRI, 1994a; EPRI, 1994b). When settlement approaches design or acceptance criteria, a re-evaluation is performed. |
| 7 | Detailed fatigue analysis using current conditions | Performed when significant cracking is found or suspected. |

effective in detecting SCC degradation in stainless steel bellows. Most degraded bellows are found during leak tests.

### 5.1.2 Plant Walkdowns

Periodic walkdown inspections of nuclear power plant areas are performed daily and before plant restart. The areas included in plant walkdowns include those accessible portions on the outside of the containment where many of the penetrations exist. The purpose of these routine walkdowns is to notice any abnormal physical conditions (such as rust, dents, cracks, deformation, discoloration, insulation damage, vibration, leaks, and spills) and any unusual noises.

Plant walkdowns are also performed after refuelings and other planned plant shutdowns and just before or during plant restart. The purpose of these walkdowns is the same as that for the daily walkdowns; however, during pre-start walkdowns, personnel must pay particular attention to areas or components that were modified, refurbished, or replaced during the outage and to general "housekeeping" conditions. These pre-start plant walkdowns are performed within the primary containment, as well as other plant areas. Therefore, all reasonably accessible portions of the penetrations are observed during these walkdowns.

All abnormal conditions identified during these walkdowns are documented and investigated and corrective action is taken as necessary. The periodic plant walkdowns are particularly useful for identifying corrosion and deformation of the mechanical penetration components. However, walkdown inspections are unlikely to detect cracks or pinholes in bellows caused by TGSCC.

### 5.1.3 Leakage Testing, 10 CFR 50, Appendix J

The primary containment, as a whole, and individual penetrations are periodically leak tested in accordance with Technical Specification requirements and the regulatory requirements of 10 CFR 50, Appendix J (1995). Although this leak testing is also part of the Surveillance Program discussed in Section 5.1.1, it is considered to be a separate program. The testing frequency can be generalized as three times during a 10-year period for the ILRT of the entire primary containment and every refueling outage for LLRTs of individual penetrations. An LLRT of a bellows pressurizes the volume between the plies and the associated test connection piping and valves, while the ILRT checks the containment and penetration as a whole. The ILRT and LLRT have specific maximum allowable leakage rates that are specified in 10 CFR 50, Appendix J, the Technical Specifications, and the test procedures. When leakage exceeds the allowable rates, an investigation of the leakage cause is performed and corrective action is taken as necessary. For bellows, corrective action generally will require replacing the entire bellows. Following the repair, a retest is performed to ensure that the bellows is leak tight.

In 1992, the NRC issued modifications to 10 CFR 50, Appendix J to provide options for performing these leakage tests. In general, this option would allow the frequencies of the ILRTs and LLRTs to be reduced based on successful experiences with prior testing and the establishment of leakage test performance goals. Although the option to reduce test frequency is, to some extent, a result of generally good performance with regard to the results of ILRTs and LLRTs over the past 20 years, it also reflects a concern for the aging effect on the primary containment and mechanical penetrations caused by frequent testing.

The LLRTs associated with mechanical penetrations that contain stainless steel expansion joints made with two-ply bellows typically have included interply leakage tests for these bellows. Now it is clear that, because of bellows manufacturing methods, this type of test is invalid and can provide nonconservative leakage rate results (Brown and Tice, 1993; Berg and Brown, 1995). The NRC notified licensees of this concern (NRC, 1992a). Complete resolution of this issue is plant-specific because not all plants have this type of two-ply bellows.

### 5.1.4 Settlement Trending Program

The Settlement Monitoring and Trending Program is identical to the programs described in the containment industry reports (EPRI, 1994a; EPRI, 1994b) to verify primary containment settlement. A similar program is used for the spent fuel pit foundation. The purpose is to check for differential settlement, which may increase loads on the fuel transfer tube penetration components.

### 5.1.5 Detailed Fatigue Analysis Using Current Condition

Fatigue potentially is a significant degradation mechanism for some components in each type of mechanical penetration. For early designs, an explicit fatigue evaluation was not required; however, if the containment is a Section III/Subsection NE design (ASME, 1992b), then fatigue analysis was required. Fatigue analyses that were performed for bellows used an undegraded component basis. This program requires an updated fatigue analysis when significant degradation is found for any of the components. The ASME B&PV Code (1992a), Section XI, In-service Inspection, includes an appendix to address fatigue when the analysis includes the calculation of cumulative fatigue usage factor. This nonmandatory appendix provides guidance for usage factors that approach the limiting value of 1.0 or other fatigue concerns. In the event that a bellows is degraded by TGSCC, however, the fatigue curves in the code are very likely not conservative. Fatigue tests on low-carbon steels with corrosion damage show a significant reduction in the number of cycles before failure (Shigley, 1983; Cherry and Smith, 2001[a]; Bruneau and Zahrai, 1997). Although the reduction in cycles caused by TGSCC has not been quantified, a similar result will certainly occur.

### 5.1.6 Cleaning

Preventive cleaning of expansion joint bellows may be performed coincident with periodic inspections or components or during related maintenance. The following substances have been found generally acceptable by various manufacturers for cleaning of penetration components. In all cases, the manufacturer's guidance should be consulted before using any substance for cleaning penetration components (Conax, 1987a[b]; Conax, 1987b[c]).

- "freon" TF (trichlorotrifluoroethane)
- "freon" TWD 602 ("freon" TF with water and detergent)
- mild detergents and soaps
- denatured alcohol ("isopropyl")

These substances generally are not recommended for use on penetrations:

- 1, 1, 2, 2 - tetrachloroethane
- 1, 1, 1 - trichloroethane (e.g., "vythene")
- acetone
- alcohols (except as noted above)
- benzene (benzol)
- carbon tetrachloride
- cellosolve solvent
- chlorobenzene
- chloroform
- cyclohexanone

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[a] Cherry, J.L. and Smith, J.A. 2001 (in press). Capacity of Steel and Concrete Containment Vessels with Corrosion Damage. Albuquerque, NM: Sandia National Laboratories.

[b] Conax Buffalo Corporation, 1987a, Proprietary Report, IPS-981, Rev. B, March 1987.

[c] Conax Buffalo Corporation, 1987b, Proprietary Report, IPS-725, Rev. H, July 1987.

- esters (e.g., ethylacetate)
- "freon 22"
- "freon" TA ("Freon" TF + acetone)
- "freon" TE ("Freon" TF + ethanol)
- heptane
- kerosene
- methylene chloride
- methyl ethyl ketone
- mineral spirits (e.g., "varson")
- most regular gasolines
- naphtha
- tetrachloroethylene ("perclene")
- toluene
- trichloroethylene (a degreasing solvent)
- turpentine
- xylene (xylol)

### 5.1.7 Repair

Degradation of bellow elements and the inability to effectively repair these components has caused the entire expansion joint to be replaced for feedwater and mainstream piping penetrations at several nuclear plants. These replacements are difficult and costly because of poor access around these large-diameter piping penetrations. Some of the replacements have longitudinal closure welds (through the expansion joint spacer and bellows areas). Without employing these expansion joint closure welds, it would have been necessary to remove an entire section of process pipe and all other penetration components to replace the expansion joint. This clearly is impractical for many physical configurations.

## 5.2 License Renewal Requirements

The current requirements for the technical content of a license renewal application are described in 10 CFR 54.21 (1995) and Nuclear Energy Institute (NEI) report NEI 95-10 (NEI, 1995). A license renewal application must contain:

1. an Integrated Plant Assessment (IPA),

2. a list of CLB changes during NRC application review,

3. an evaluation of time-limited aging analyses (TLAAs), and

4. an FSAR supplement.

An IPA must

1. for those SSCs within the scope delineated in 10 CFR 54.4 (1996), identify and list those structures and components subject to AMR;

2. describe and justify the methods used in item 1 (scope determination) of the IPA; and

3. for each structure and component identified in item 1 of the IPA, demonstrate that the effects of aging will be managed so that the intended function(s) will be maintained for the period of extended operation.

### 5.2.1 Aging Management Review

The revised version of the license renewal rule, (10 CFR 54, 1996), discusses the concept of an "aging management review." The IPA must list structures and components subject to an AMR, and must demonstrate that aging concerns are being managed. It is necessary to maintain the CLB in a way so that there is an acceptable level of safety during the period of extended operation. Furthermore, the TLAAs required for some penetrations, (10 CFR 50.49, 1983; 10 CFR 54.21, 1995) which form the basis of a plant operator's conclusion regarding the capability of those penetrations, must consider the effects of aging and be based on explicit assumptions defined by the current operating term of the plant. The specific methods to be used by the plant operator in meeting these requirements are not mandated; each operator can select the method(s) or technique(s) most appropriate for their individual AMR.

The 10 CFR 54 (1996) requires that the IPA list those structures and components subject to an AMR and demonstrate that the effects of aging on the functionality of such structures and components will be managed to maintain the CLB so that there is an acceptable level of safety during the period of extended operation. Furthermore, the TLAAs required by 10 CFR 54.21 (1995), which form the basis of a licensee's conclusion regarding the capability of SSCs, must consider the effects of aging and be based on explicit assumptions defined by the current operating term of the plant. The methods by which the licensee meets these requirements may vary; each licensee can select the method(s) or technique(s) most appropriate for their individual AMR.

### 5.2.2 Special Considerations for Stainless Steel Bellows

The problems with interply leak testing of two-ply stainless steel bellows were identified in NRC Information Notice 92-20 (1992). This issue has been explored in greater detail by Brown and Tice (1993) and Berg and Brown (1995). The interply pressurization method for leak testing two-ply bellows provides invalid and nonconservative results because the air path between the bellows from one end to the other and around the circumference is essentially blocked by the stainless steel mesh between the plies. The function of the mesh was to hold the spacing between the two plies; however, it also acts as a gas seal, at least for an extended period of time. To perform a valid leakage test across the expansion joint bellows, a pressure must be applied to the exterior or interior surface and the level of pressure decay measured over a period of an hour or two. This test can be accomplished with a permanent modification to the penetration. A second bellows expansion joint is placed outside the existing joint, and the leakage test is performed by pressurizing the area between the primary and secondary (or outer) expansion joint. Localized permanent test enclosures around bellows can be designed and used for leak testing as described by Brown and Tice (1993) and Berg and Brown (1995).

Valid LLRTs for these two-ply bellows expansion joint penetrations, as well as single-ply bellows, can be accomplished with a temporary test fixture. This fixture, installed inside the containment, would result in testing all pressure-retaining penetration components, including the expansion joint, simultaneously. Although it obviously is possible to build such fixtures for testing as noted by Brown and Tice (1993) and Berg and Brown (1995), it is time consuming and expensive, especially at plants with approximately 20 penetrations of this type for certain containment designs. A reusable test assembly that can be installed and removed rapidly would be preferred for periodic penetration LLRTs. Where space does not allow use of local bellows test assemblies, the replacement of a close-rolled two-ply bellows with two independent, single-ply bellows is one of the few alternatives remaining to satisfy the LLRT requirements of 10 CFR 50, Appendix J (1995).

# 6. REGULATORY AND RISK SIGNIFICANCE OF DEGRADED BELLOWS

This section explores the potential regulatory and risk significance of degraded bellows in the context of regulatory *requirements* for DBA and in the context of risk *goals* related to SAs. The primary focus is on BWRs with Mark I containments because they typically have a large number of two-ply bellows associated with containment penetrations and a single-ply bellows associated with each of the vent line pipes. However, the insights developed here for Mark I containments can be adapted to other containments that employ bellows.

The section first examines the decision logic, representative of at least some part of the industry, relating to the identification and replacement of two-ply bellows. Insights from SNL's containment bellows testing program are then reviewed in the context of industry logic for bellows replacement. Finally, a number of issues related to the risk significance of containment bellows are reviewed.

## 6.1 Representative Industry Logic for Bellows Replacement

In typical installations of new bellows assemblies, initial LLRT values are zero or very small (Figure 6-1). Trending of LLRT data at some domestic BWR plants indicates that the measured leak rate remains stable for a period of time, suggesting that the condition of the bellows has remained essentially unchanged. After a period of 5 to 15 years, an increase in leakage rates has been observed at some drywell penetrations. Initially these leak rates tend to increase linearly with time, indicating that whatever is causing the rates to change is progressive.

Figure 6-2 is a simplified logic diagram (Brown and Tice, 1993) for replacement of penetration bellows. Implicit in this logic is an assumption that an interply LLRT is an effective leak detection method while acknowledging that an interply LLRT is not a reliable *leak rate* measurement method. It is clear from Figure 6-1 that the onset of leakage can be detected when the measured leak rate exceeds some threshold value, although Brown and Tice (1993) do not suggest a threshold value. However, examination of Figure 6-1 suggests that a measured leak rate of ~0.05 $m^3/hr$ (1.20 $m^3/day$) might be adequate for detecting departure from a stable state.

Figure 6-2 shows that the bellows may be judged to be an acceptable pressure boundary even if one or both plies are known to be degraded and leaking. When both plies are degraded, all observable flaws in the outer ply are mapped and measured as input to a calculational methodology, which involves predicting the gas flow rate through crack-like orifices. In an effort to account for radial expansion from pressure (which causes the cracks to open slightly), a finite element model is used to simulate the bellows response to pressure loads. Empirical adjustments are made to reflect known flow rates through known flaws. The calculational methodology is used to *predict* bellows response during a DBA, and as such, is the foundation on which a decision is made to replace (or defer replacement).

The logic in Figure 6-2 is applied to each penetration bellows separately. No adjustments are made to situations where multiple bellows are simultaneously in a degraded state. Such common mode or concurrence states likely existed at Dresden where a number of bellows were replaced over two consecutive refueling outages.

## 6.2 Sandia National Laboratories Testing of Like-New and Degraded Bellows

SNL has tested containment piping bellows that were in like-new condition (Lambert and Parks, 1994) and in degraded condition (Lambert and Parks, 1995). All bellows were conducted to full compression (or extension) with an appropriate lateral displacement. Table 6-1 summarizes the tests conducted with like-new bellows subjected to SA conditions. These tests spanned conditions ranging from ambient pressure and temperature to elevated pressure and temperature. Table 6-2 summarizes the tests conducted with degraded bellows. The degraded bellows were subjected to accelerated corrosion environments followed by combinations of elevated pressure, temperature, and axial (compressive) and lateral deformations.

Table 6-3 summarizes key insights from SNL's testing of containment bellows. Under DBA conditions, the expected deformation on any individual bellows is ~10% of full compression (or extension). Under DBA conditions, SNL's experiments showed no leakage from

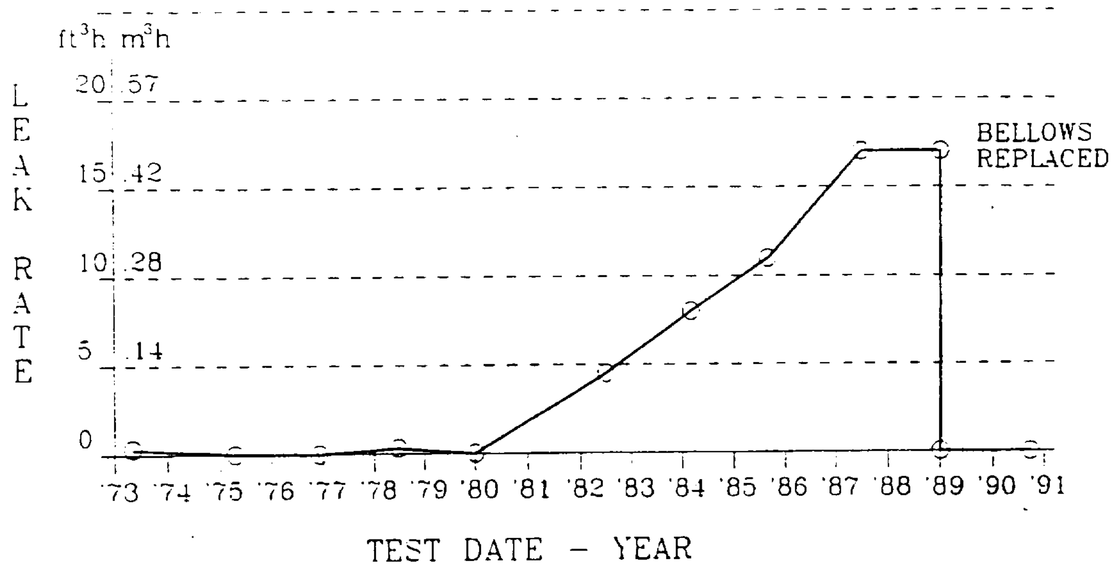


**Figure 6-1. LLRT Flow Rate vs. Time (from Brown and Tice, 1993).**

like-new bellows. Also, the bellows with laboratory-induced TGSCC (degradation not extensive enough to cause an initial leak), did not leak when compressed (or extended) about 10% to correspond to a DBA.

These three categories of tests covered in Table 6-3 demonstrate that if a bellows with TGSCC does not leak initially, then the relative movement caused by a DBA *probably* would not cause a leak to develop. However, only three of the bellows tested that had TGSCC damage did not leak initially. Obviously, three tests do not comprise a statistical sample. Probably the most important factors are the crack growth rate and the associated increase in the leak rate as the bellows in the test were displaced. Up to about 10 or 20% compression, the leak growth rate increased, but at a stable rate. Therefore, if a bellows was not leaking initially, a design DBA would be unlikely to cause a leak. In the unlikely event that a leak did develop, it can be stated with very high confidence that the leak rate would be small enough to pass the 10 CFR 50, Appendix J (1995) leak rate criteria. Consequently, DBA criteria are demonstrably satisfied.

No such conclusions can be drawn when the bellows is corroded to the point that they leak initially (before being compressed or elongated). The data indicate that the initial leaks increase, but the measurement techniques are not adequate to quantify if DBA criteria are satisfied. The database also is not adequate to validate industry models that predict bellows performance under DBA conditions from a prior characterization of damage with the bellows in the neutral position.

Table 6-3 shows that any bellows, like-new or degraded, can develop leaks for SA where the bellows are completely compressed (or extended). The leaks take the form of cracks, localized circumferential tears, or tears where a convolution buckles. The welded seam appears to be a preferred site for buckle tears. Leakage increases with the degree of initial degradation to the bellows. Any bellows failure at full compression would generally be classified as a small leak (~10 times DBA criteria). The IPA treatment of containment fragility suggests that containment deformations capable of fully compressing a bellows will have a significant likelihood of inducing other failures that will produce much larger hole sizes. Table 6-1 also suggests that expansion bellows may develop more risk-significant leaks when fully extended. The available data do not comprise a statistical sample, and there are no data for degraded expansion bellows.

The pressure history on a Mark I drywell has a similar response for both DBA and SA. The drywell pressure rises rapidly during the duration of the blowdown event (minutes) and then quickly decays. The magnitude of the pressure rise is controlled by the rate of blowdown (in competition with flow through the vent lines into the wetwell); consequently, the pressure rise could be larger in a SA where the lower ruptures with the system at pressure. For both DBA and SA, the drywell pressure will again increase slowly over a period of several hours.

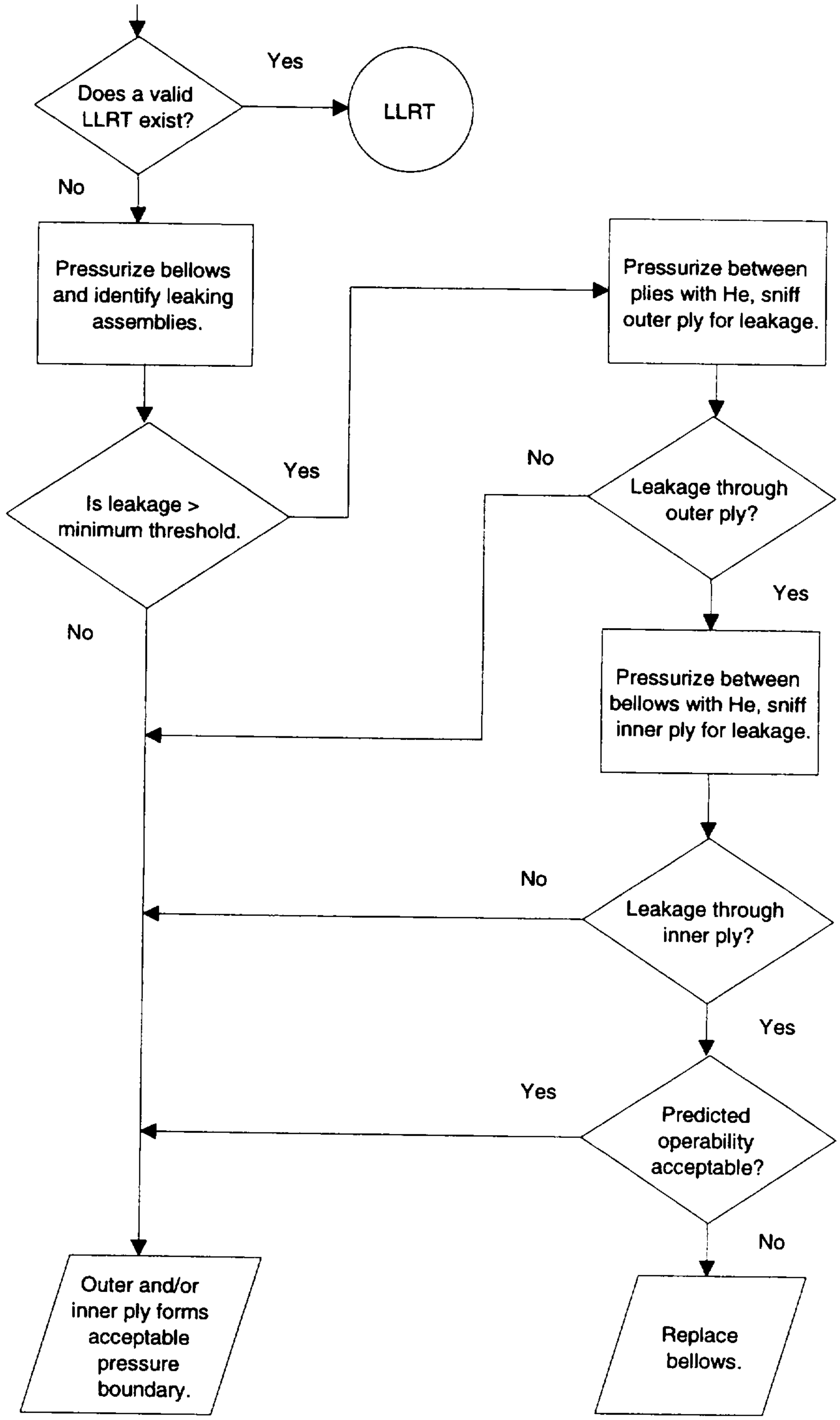


**Figure 6-2. Representative of Industry Logic for Replacement of Bellows.**

**Table 6-1. Experiment Results from Like-New Containment Piping Bellows Subjected to Severe Accident Conditions**

| Test ID[(1)]<br>Specimen ID[(2)] | Direction | Pressure Temperature | Leak Tight at Full Compression (Elongation) of First Loading (Displacement of First Leak If It Occurs) |
|---|---|---|---|
| I-11-CL<br>A-8-2-1 | Compression | Ambient pressure<br>Ambient temperature | Leak tight |
| I-12-LC<br>A-8-2-2 | Compression | Ambient pressure<br>Ambient temperature | Leak tight |
| I-3-SC<br>B-8-2-1 | Compression | 0.52 MPa<br>Ambient temperature | Leak tight |
| I-4-SCT<br>B-8-2-2 | Compression | 0.45 MPa<br>213°C | Leak at 86% compression |
| I-2-SCT<br>B-12-2-1 | Compression | 1.03 MPa<br>216°C | Leak tight |
| I-10-SCT<br>B-12-2-1 | Compression | 0.47 MPa<br>210°C | Leak at 91% compression |
| I-9-SC<br>B-12-2-2 | Compression | 0.50 MPa<br>Ambient temperature | Leak at 99% compression |
| II-19<br>VL-2 | Compression | 1.03 MPa<br>213°C | Leak tight |
| II-17<br>PP-2 | Compression | 0.51 MPa<br>204°C | Leak tight |
| I-13-SE<br>A-8-1-e | Elongation | Ambient pressure<br>Ambient temperature | Leak at 88% elongation |
| I-14-SE<br>A-8-2-e | Elongation | Ambient pressure<br>Ambient temperature | Leak tight |
| I-15-SE<br>B-8-1-e | Elongation | Ambient pressure<br>Ambient temperature | Leak tight |
| I-16-SE<br>B-8-2-e | Elongation | Ambient pressure<br>Ambient temperature | Leak tight |

(1) Test ID Description

CL Axially compress the bellows until all convolution roots are fully compressed. If the bellows are still leak tight, apply lateral deformation until a tear in the bellow occurs. No internal pressure will be applied during this test.

LC Apply lateral offset until the bellows becomes unstable (i.e., very small increase in lateral load produces relatively large lateral deformation). If the bellows is still intact, apply axial compression until all convolutions are fully compressed. If the bellows is still intact after being fully compressed, apply additional lateral deformation until a tear develops in the bellows. No internal pressure will be applied during this test.

SC Simultaneously apply internal pressure, axial compression, and lateral deformation. Internal pressure, axial compression, and lateral deformation shall be increased linearly so that the internal pressure reaches the maximum test pressure level when the total applied lateral deformation is 2 inches when the bellows are fully compressed. If the bellows are still intact after being fully compressed, reverse the applied test conditions by removing axial and lateral deformation, as well as internal pressure at the same rate at which each was originally applied. Continue unloading until either a crack develops or until all originally applied displacement and pressure have been removed.

SCT Same as SC except test temperature is 425°F (±25°F) throughout the specimen for the entire test.

SE Simultaneously apply axial elongation and lateral deformation. Axial elongation and lateral deformation shall be increased linearly so that total applied lateral deformation is 2 inches when the bellows is fully elongated. If the bellows are still leak tight after being fully elongated, reverse the applied test conditions by removing elongation and lateral deformation at the same rate at which each was originally applied. Continue this process until either a crack develops or until all originally applied deflection has been removed. No internal pressure will be applied during this test.

(2) Specimen ID numbers denote the bellows construction. The first letter (A or B) indicates convolution depth (A = 0.5 in. and B= 1.25 in.). The first number indicates the number of convolutions (8 or 12), and the second number indicates the number of plies. Specimens with A or B designation are single bellows; those with VL (vent line) and PP (process piping) are universal bellows.

**Table 6-2. Experiment Results from Degraded Bellows Subjected to Severe Accident Conditions**

| Test ID[1]<br>Specimen ID[2] | Surface Condition<br>Pretest Leak | Compressive Phase Observations | Measured Interply<br>Leak Rates |
|---|---|---|---|
| I-7-SC<br>A-12-2-1 | • Severely corroded with numerous small cracks and a few larger ones.<br>• Pretest leaks > App J | • Pretest cracks began to grow at 10% compression.<br>• Cracks merged and expanded at 25% compression.<br>• Severe damage to the bellows with numerous large tears at full compression. Wire mesh clearly visible. Severe cracking in other places. Many failures along welded seam.<br>• Estimated flow area from archival photos $\sim 1.4\times10^{-3}$ m²<br>• Estimated volume flow rate at 293 K<br>$\dot{V} = A\sqrt{\gamma RT} = 1686$ m³/day | |
| I-8-SCT<br>A-12-2-2 | • Corrosion visible to naked eye, but cracks only visible only with magnification.<br>• Pretest leaks < App J<br>• Pretest leak ~0.12 slpm ~$8\times10^{-8}$ m³/day | • Leak rate increased with additional deformation from ~25% compression until 50–75% compression.<br>• Wire mesh clearly visible in a number of locations. Cracking in other places.<br>• Estimated flow area from archival photos ~$2.2\times10^{-4}$ m²<br>• Estimated volume flow rate at 293 K<br>$\dot{V} = A\sqrt{\gamma RT} = 265$ m³/day | Compression Leak Rate (sm³d) @0.14 MPa<br>0.00 0.12<br>0.21 0.40<br>0.26 1.87<br>0.31 5.83<br>0.36 12.2<br>0.41 34.7<br>1.00 39.6<br>Max leak 73.4 sm³d after return to full extension |
| I-5-SCT<br>C-6-2-1 | • Well-defined cracks easily visible.<br>• Pretest leak < App J<br>• Pretest leak ~0.005 slpm<br>• Pretest leak ~0.12 slpm ~$0.3\times10^{-8}$ m³/day | • Leak rate increased with additional deformation from ~25% compression until 50-75% compression.<br>• Wire mesh clearly visible in a few locations.<br>• Estimated flow area from archival photos ~$2.2\times10^{-4}$ m²<br>• Estimated volume flow rate at 293 K<br>$\dot{V} = A\sqrt{\gamma RT} = 265 m^3/day$ | Compression Leak Rate (sm³d) @0.14 MPa<br>0.00 0.01<br>0.11 0.04<br>0.21 0.13<br>0.35 0.40<br>0.40 1.68<br>0.45 7.08<br>0.55 12.1<br>0.61 24.5<br>0.64 41.6<br>0.76 54.6<br>1.00 54.6<br>Extension after full compression<br>Compression Leak Rate (sm³d) @0.14 MPa<br>0.94 59.1<br>0.60 83.6 |
| I-1-SCT<br>A-8-2-4 | • Visible corrosion with minute cracks visible with magnification.<br>• No pretest leak. | • Small leak at last increment to full compression. | Compression Leak Rate (sm³d) @0.14 MPa<br>0.00 0.00<br>1.00 0.18<br>Extension after full comp.<br>Compression Leak Rate (sm³d) @0.14 MPa<br>0.85<br>0.71 24.5 |

**Table 6-2. Experiment Results from Degraded Bellows Subjected to Severe Accident Conditions (continued)**

| Test ID[1] Specimen ID[2] | Surface Condition Pretest Leak | Compressive Phase Observations | Measured Interply Leak Rates |
|---|---|---|---|
| II-20-SCT<br>VL-1<br>Single-ply<br>Vent line bellow | • Visible corrosion with minute cracks visible with magnification.<br>• No pretest leak. | • No leak to full compression.<br>• No leak for two full cycles.<br>• Cracks became wider without leaking by end of test. | |
| I-6-SCT<br>D-6-2 | • Pretest pitting but no visible surface corrosion or cracking.<br>• No pretest leak. | • No leak prior to full compression. | |

Notes:
(1) Same as Table 6-1.
(2) Specimen ID numbers denote the bellows construction. The first letter (A, C, or D) indicates convolution depth (A and C = 0.5 in. and D=1.25 in.). The first number indicates the number of convolutions (6, 8, or 12), and the second number indicates the number of plies. Specimens with A designation are single bellows; those with C, D, or VL (vent line) are universal bellows.

**Table 6-3. Insight from Sandia National Laboratories' Testing of Like-New and Degraded Containment Bellows**

| | Like-New Bellows and Degraded Bellows Not Leaking Initially | Degraded Bellows Initial Leak Rate < Appendix J Criteria* | Degraded Bellows Initial Leak Rate > Appendix J Criteria* |
|---|---|---|---|
| DBA Conditions < 10% Compression | • No measurable LLRT leaks.<br>• DBA criteria demonstrably achieved. | • Interply leak rate measurements (LLRT) suggest that small deformations increased the leak rate.<br>• Not possible to quantify if DBA criteria are met (ILRT leak rates cannot be determined from the LLRT). | • Interply leak rate measurements (LLRT) suggest that the small deformations increased the leak rate.<br>• Not possible to quantify if DBA criteria are met (ILRT leak rates cannot be determined from the LLRT). |
| | • Industry practice leaves bellows in service. | • Industry practice is to leave bellows in service if projected leak rate through either ply is less than 10 CFR 50, Appendix J (1995) criteria. | • Industry practice is to leave bellows in service if projected leak rate through either ply is less than 10 CR 50 Appendix J (1995) criteria.<br>• Often, bellows must be replaced. |
| Severe Accident Conditions Up To 100% Compression | • Possible tears initiate near full compression or extension.<br>• Possible leaks develop near full compression or extension, with larger leak rates in expansion bellows than in compression bellows. | • Leak rates increase as bellows is compressed.<br>• Estimated leak rates ~12 times DBA criteria when bellows is fully compressed (based on a small number of test results). | • Leak rates increase as bellows is compressed.<br>• Estimated leak rates ~80 times DBA criteria when bellows is fully compressed (based on a small number of test results). |

* 10 CFR 50, 1995

This behavior is significant with regard to potential radionuclide release from the containment for two reasons. First, any leaks in the containment bellows are insignificant compared to the flow into the wetwell where the fission products will be scrubbed. Second, the period when short-term drywell pressures are high is limited, which further reduces the release from the containment. Furthermore, for the more common sequences where there is power in the plant, the operators have the means to vent the wetwell to reduce long-term pressures in the drywell, reducing even further any possible fission product release. SNL also notes that the core is not severely damaged in a DBA, so that the fission product release to the drywell is also quite limited.

## 6.3 Severe Accident Risk Perspective Regarding Bellows Leakage

A number of probabilistic risk assessments (PRA) have been performed that have quantified the potential risk from various accident scenarios. In particular, the NUREG/CR-1150 (NRC, 1990) study has been reviewed to determine what information from this previous study can be applied to the bellows leakage issues. Since bellows are used primarily in steel containments, the current discussion will focus on the two NUREG/CR-1150 containments that are of steel construction, the Peach Bottom plant (BWR) and the Sequoyah plant (PWR) (Payne et al., 1990; Gregory et al., 1990, respectively).

Bellows leakage is addressed in terms of containment failures, source terms, and consequences that were included in the NUREG/CR-1150 study (NRC, 1990). Three primary aspects of the release should be addressed

- size of the leak,
- source term (which radionuclides and what quantities) to the environment, and
- timing characteristics associated with the release.

If a bellows leak were to occur because of TGSCC damage, the leak will generally be one or more line cracks that form in the convolutions of the bellows. In terms of the NUREG/CR-1150 PRA (NRC, 1990), the size of a bellows leak would probably lie somewhere between nominal leakage, applied in the NUREG/CR-1150 study during times of no containment failure (NCF), and a small leak. As applied in NUREG/CR-1150, the NCFs nominal leakage was estimated at 0.1% per day for five days, and the small leak was estimated as a 1 $ft^2$ hole. Based on the limited data available, it appears that any leak through a bellows would be much closer to the nominal leakage category than it would be to the small leak category. In any event, it is assumed that the NUREG/CR-1150 small leak (1 $ft^2$ hole) is an upper bound, with a large margin for error, for a worst-case bellows leak.

The magnitude of the source term release is dependent on many factors, including:

- the accident initiator;
- the amount of material released from the reactor core;
- the core release that is retained in vessel;
- the vessel release that is retained in containment; and
- the decontamination by mitigating factors such as suppression pools, sprays, ice condensers, natural agglomeration, etc.

It is beyond the scope of this discussion to investigate all of these features of the source term. The feature that is central to the bellows leakage issue involves the containment retention, which can be investigated by focusing on containment failure size and timing. The remaining factors are indirectly included in the distribution of the source terms within the range that is investigated herein.

For timing considerations, because a bellows failure could exist throughout the entire accident progression, source terms for early containment leaks (ECLs) are considered. In addition, to attempt to account for smaller leaks (with lower leak rates), source terms that involve late containment leaks (LCLs) are considered because they feature longer containment retention times. Finally, source terms involving NSF that were modeled in NUREG/CR-1150 (NRC, 1990) with the 1% per day leakage rate are considered. Note that all source terms discussed here will include only those core damage accidents that proceed to vessel breach (there were some accidents modeled in NUREG/CR-1150 in which core damage was arrested in vessel). As mentioned earlier, a bellows failure could exist throughout the entire accident progression. Hence, it is important to look at emergency response scenarios in which the population evacuates after the release has begun.

The NUREG/CR-1150 PRA (NRC, 1990) implemented a partitioning method in which the thousands of source terms estimated in each plant analysis were placed into source term groups based on their potential for early and latent health effects. Consequence calculations were then performed for the frequency-weighted average source term for each group and each of three subgroups. Subdivision into the three subgroups was based on evacuation timing:

Subgroup 1: evacuation starts at least 30 minutes before the release begins,

Subgroup 2: evacuation starts between 30 minutes before and 1 hour after the release begins,

Subgroup 3: evacuation starts more than 1 hour after the release begins.

In NUREG/CR-1150 (NRC, 1990), source terms for some subgroups were not calculated because the subgroup had zero probability. Therefore, for this study, the source term subgroups are selected on the basis of evacuation timing in the following order: Subgroup 2, Subgroup 3, and Subgroup 1. The source term subgroup designation is the final number in the source term group descriptor. (For example, the source term group PB-XX-1 is Peach Bottom source term group XX, Subgroup 1, and the source term group SEQ-XX-2 is the Sequoyah source term group XX, subgroup 2.)

Based on the selection criteria of containment failure leak size, containment failure time, and evacuation timing, relevant source term groups from the Peach Bottom and Sequoyah NUREG/CR-1150 (NRC, 1990) analyses have been chosen to establish the discussion of the bellows leakage event. These source term groups are the most probable for the containment failure conditions noted. For Peach Bottom, this results in the selection of the following source term groups:

1. For ECL failures, the most probable source term groups are PB-05 and PB-01 (subgroups PB-05-3, PB-01-3).

2. For LCL failures, the most probable source term groups are PB-19 and PB-01 (subgroups PB-19-3, PB-01-3).

3. For NCF, the most probable (as well as the only) source term groups are PB-18 and PB-17 (subgroups PB-18-1, PB-17-1).

For Sequoyah, these are the selected source term groups:

1. For ECL failures, the most probable source term groups are SEQ-18 and SEQ-02 (subgroups SEQ-18-2, SEQ-02-2).

2. For LCL failures, the most probable (as well as the only) source term groups are SEQ-18 and SEQ-17 (subgroups SEQ-18-2, SEQ-17-2).

3. For NCF, the most probable (as well as the only) source term groups are SEQ-17 and SEQ-16 (subgroups SEQ-17-2, SEQ-16-2).

The consequences calculated in NUREG/CR-1150 (NRC, 1990) for the selected source terms are provided in Table 6-4. The consequences are conditional on the occurrence of each source term group. Hence, to obtain an indication of risk, each consequence must be multiplied by the estimated frequency of occurrence of the source term group (e.g., the frequency of a SA coincident with a bellows leak). Also note that these results are site-specific with respect to such considerations as population density and distribution, meteorology, core size, etc. The results indicate that the overall predicted consequences are moderate, in terms of the entire SA spectrum of both early and latent health effect consequences, for the worst case surrogate scenario of ECL for both Peach Bottom and Sequoyah.

In most PRAs, the source term and consequence analyses typically focus on accidents that will result in the highest risk (the product of a consequence and its probability). Hence, analytical resources are usually directed to events that result in either high or moderate consequences that have some non-zero probability of occurrence. Conversely, the modeling of low consequence events (such as for source term groups PB-17 and SEQ-16) is usually performed in a manner that is limited in detail, regardless of the probability of occurrence.

Even though there may have been limited resources directed toward the calculation of the source terms and consequences for some of the source term groups, it is believed that this discussion forms a valid preliminary study of the potential consequences from bellows leak failure. In addition, it is recommended that because this study indicates a potential for both early and latent health effects, that from a risk perspective, the bellows leak failure should be further investigated.

Table 6-4. Consequences for Selected Source Terms from NUREG/CR-1150 (NRC, 1990)

| Source Term Subgroup | Containment Failure Conditions | Early Fatalities | Latent Cancer Fatalities | Population Dose within 50 miles (Sv) | Population Dose Entire Region (Sv) | Individual Early Fatalities Risk, 1 mi. | Individual Late Cancer Fatalities Risk, 10 mi. |
|---|---|---|---|---|---|---|---|
| PB-01-3 | ECL,LCL | 9.00E–7 | 3.94E+1 | 1.32E+3 | 2.50E+3 | 3.25E–9 | 7.39E–5 |
| PB-05-3 | ECL* | 1.87E–2 | 5.36E+2 | 1.86E+4 | 3.79E+4 | 5.85E–5 | 1.34E–4 |
| PB-17-1 | NCF | 0.00E+0 | 1.33E–2 | 5.22E–1 | 9.28E–1 | 0.00E+0 | 3.98E–9 |
| PB-18-1 | NCF* | 0.00E+0 | 7.21E–1 | 3.22E+1 | 5.71E+1 | 0.00E+0 | 4.78E–7 |
| PB-19-3 | LCL* | 0.00E+0 | 5.48E+1 | 1.74E+3 | 3.28E+3 | 0.00E+0 | 8.88E–5 |
| SEQ-02-2 | ECL | 7.24E–5 | 6.09E+1 | 1.26E+3 | 3.78E+3 | 1.82E–7 | 8.12E–5 |
| SEQ-16-2 | NCF | 0.00E+0 | 6.02E–1 | 1.98E+1 | 3.21E+1 | 0.00E+0 | 3.03E–6 |
| SEQ-17-2 | LCL,NCF* | 0.00E+0 | 2.53E+0 | 8.09E+1 | 1.82E+2 | 0.00E+0 | 7.71E–6 |
| SEQ-18-2 | ECL,LCL* | 0.00E+0 | 1.84E+2 | 3.06E+3 | 1.05E+4 | 0.00E+0 | 1.34E–4 |

* Denotes most probable source term group for this containment failure condition.

## 6.4 Future Research Recommendations

The previous discussion suggests that the following activities would form the basis for a comprehensive and defensible resolution to the degraded bellows issues.

1. There is a need to *quantify* the leak rate from degraded bellows, with measurable leak rates in excess of a threshold value, under DBA conditions (less than approximately 10% compression or expansion). To quantify the leak rates, the experiments should be performed with single-ply bellows at temperatures and pressures representative of DBAs.

2. These experiments can be used to benchmark industry models for predicting leak rates under DBAs given attempts to characterize the damage state under neutral conditions. This activity requires coordination and cooperation with industry for maximum benefit.

3. Calculations using a system level code, such as MELCOR, should be performed using realistic leak rates for DBA and SA conditions. The calculations should also take credit for realistic pressure transients, given the passive design features and accident management procedures that are likely to be invoked during an accident. These elements can be integrated into a probabilistic framework to give a balanced risk informed perspective to the impact of degraded bellows.

The following two tasks would also provide useful information, but are lower in priority than the first three listed above. However, if DBA leak rate testing is performed, the incremental cost to do one of the two tests that follow would be minimal. Because both of these tests cause the bellows to fail, only one or the other could be performed using the bellows from the previous DBA test.

a. Cycle the degraded bellows under DBA conditions (i.e., an earthquake). For example, compress the bellows to 10% compression, elongate to the neutral position, compress to 10%, elongate to neutral, etc., and monitor the leak rate as TGSCC cracks develop and grow.

OR

b. *Quantify* the leak rate from degraded bellows from a single SA excursion (i.e., 100% compression).

# 7. SUMMARY AND CONCLUSIONS

This report has presented a comprehensive examination of currently available information regarding bellows used in containment penetrations, materials, the conditions that cause aging, applicable aging mechanisms, and aging effects, and the currently available methods that can be used to monitor aging.

## 7.1 Summary

1. The aging of bellows is routinely assessed and monitored through a combination of leak rate testing and surveillance programs.

2. The primary containment, as a whole, and individual bellows are periodically leak tested in accordance with Technical Specification requirements and the regulatory requirements of 10 CFR 50, Appendix J (1995). When leakage exceeds the allowable rates, an investigation of the leakage cause is performed and the bellows is replaced if the projected leak rate (for the next outage) is greater than Appendix J criteria allow. Usually an immediate retest is performed to ensure that corrective action was appropriate.

3. The Type B LLRTs associated with penetrations that contain stainless steel expansion joints made with two-ply bellows typically have included interply leakage tests for these bellows. It is now clear that this type of test is invalid and can provide nonconservative leakage rate results.

4. Plants are not required to replace a bellows until the leak rate is large enough to cause the penetration to fail either the LLRT or ILRT.

5. Routine daily plant walkdowns are conducted in all plant areas, except within primary containment, to detect any abnormal physical conditions such as rust, dents, cracks, deformation, discoloration, insulation damage, vibration, leaks, spills, or any unusual noises. The accessible portions of mechanical penetrations outside of primary containment are observed during these walkdowns.

6. Plant walkdowns of all areas are performed after refueling and other planned plant shutdowns and just before plant restart (in this case, personnel pay particular attention to areas or components that were modified, refurbished, or replaced during the outage). All reasonably accessible portions of the penetrations are observed during these walkdowns. All abnormal conditions identified during these walkdowns are documented and investigated and corrective action is taken as necessary. Although many kinds of damage can be detected during walkdowns, it is unlikely that TGSCC will be observed during a walkdown.

7. Degradation of bellows and the inability to effectively repair them has caused the entire expansion joint to be replaced for piping penetrations at several nuclear plants. These replacements are difficult and costly because of poor access around the large diameter penetrations.

8. A structured penetration inspection program based on plant-specific operating experience and coupled with root cause analysis and appropriate corrective action could be considered an aging management program. Note that many existing plant activities not specifically focused on penetration aging management may often be credited with addressing aging effects.

9. Effective July 1996, plants have implemented the Maintenance Rule (10 CFR 50.65, 1996). The investigation of MPFFs and the actual monitoring of penetration performance will support aging management for bellows.

## 7.2 Conclusions

The following conclusions regarding bellows historical performance can be supported:

1. Bellows have primarily failed because of TGSCC.

2. Degradation of bellows has been identified primarily through integrated and LLRTs.

3. TGSCC of stainless steel bellows is not easily detectable using visual inspection techniques. After leak tests have indicated leaks in a penetration, PT tests have been used to examine the bellows. PT can only identify damage that is on the outside of the accessible surface. It is not clear how significant the damage must be before it can be observed using PT.

4. Given an aggressive environment (i.e., a humid environment and nearby insulation with chlorides, such that chlorides leach out of the insulation and onto the bellows), bellows failure can occur over a

period of a few years. Bellows failures have occurred in plants that are fairly new, as well as in older plants. New bellows that have replaced older bellows have also developed TGSCC, and had to be replaced themselves. On the other hand, many older bellows have not experienced TGSCC. Therefore, it appears that the environment (i.e., presence of chlorides or other chemical agents) is a major factor, and bellows age, by itself, is not a good indication of the likelihood of TGSCC.

5. No aging mechanisms other than those identified in this report are known or expected to occur during this period (i.e., no mechanisms unique to the license renewal period have been identified).

6. Aging and degradation mechanisms of stainless steel expansion joint bellows are not fully controlled through current maintenance, surveillance, and condition monitoring techniques employed by operating plants. This is from the lack of any reliable means for detection of TGSCC.

7. Plants are not required to replace a bellows until the leak rate is large enough to cause the penetration to fail either the LLRT or ILRT. However, testing (Lambert and Parks, 1995) has shown that when subjected to large displacements, bellows with TGSCC damage develop leaks, or develop larger leaks, than would occur for a bellows without TGSCC.

8. Once a bellows ply develops cracks large enough to cause it to fail a leak test, there is no way to repair it and still maintain the design characteristics required. The bellows must be replaced, although this is relatively difficult and costly.

9. Bellows components require little in the way of periodic maintenance.

## 7.3 Additional Research Needed

1. The capability to determine the aging condition of bellows using a nondestructive test would be useful. Currently, the detection of SCC is extremely difficult, and evaluating the baseline condition is limited to "leak" or "no leak" determinations. Detecting damage, and quantifying the extent of the damage, is important for the long-term reliability of bellows.

2. It is also important to understand how TGSCC damage affects leakage through the bellows during both design and SA scenarios. For example, a bellows degraded by TGSCC may be able to pass a leak test (i.e., leakage is less than allowable limit), if relative movement is minimal, but cracks and leak rate could increase significantly during an accident that required significant movements be accommodated by the bellows. Testing to quantify the leak rate under DBA conditions would either verify that current industry practice (i.e., allowing leaking bellows to remain in service if the leak is small enough) is acceptable, or provide information about how large the leak rate could become during the DBA. (See Section 6.3 for additional recommended leak rate testing.)

3. Calculations using a system level code, such as MELCOR, could be performed using realistic leak rates for DBA and SA conditions. The calculations could take credit for realistic pressure transients, given the passive design features and accident management procedures that are likely to be invoked during an accident. These elements can be integrated into a probabilistic framework to give a balanced risk informed perspective to the impact of degraded bellows.

4. Are replacement bellows as tough as the original bellows were? Some bellows have been replaced that consist of two halves, with longitudinal welds that join the two parts. Can these "new" bellows withstand, without leaking, the same amount of relative movement that the original bellows could when it was new? The principal concern is the difficulty of making a longitudinal weld along the convolutions of the bellows, coupled with very large strains that occur in the bellows convolutions. Limited testing could evaluate whether the "new" bellows are as robust as the "old" bellows.

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Herman L. Graves III, NRC Project Manager

11. ABSTRACT *(200 words or less)*


In commercial nuclear power plants, many containments use stainless steel bellows at piping penetrations, in dry-well-to -wetwell vent lines, or as a part of the fuel transfer tube. Bellows are an integral part of the primary containment pressure boundary in nuclear power plants. These bellows allow for thermal expansion and contraction of the pipe, vent line or fuel transfer tube, and accommodate relative motion between the containment and other structures, such as a shield wall.

Ensuring that bellows in operational plants remain leak tight is an important safety issue. A number of bellows have been replaced when transgranular stress corrosion cracking caused them to leak. Determining how to evaluate bellows for age-related damage is a critical issue that must be addressed for plants operating under their original license, as well as for plants applying for license extension.

The types of bellows, locations in a containment, and required bellows leak tests are described for major categories of boiling-water reactor (BWR) and pressurized-water reactor (PWR) power plants. Compilation of instances of bellows cracking, excessive leakage, or structural integrity questions, as well as testing issues and other degradation problems, are reviewed and summarized. The aging mechanisms that caused the degradation are discussed. Inspection methods currently in use, and a typical industry approach to managing bellows aging are described. Degradation that resulted from corrosion, misalignment, and other damage was considered. The risk and regulatory significance of containment bellows when subjected to Current License Basis (CLB), degraded, and severe accident (SA) conditions are also addressed.


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