Transactions of the **14th International Conference on Structural Mechanics in Reactor Technology** (SMiRT 14), Lyon, France, August 17-22, 1997

D01/4

Technical information from industry reports addressing license renewal

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ABSTRACT

The USNRC reviewed and documented, in NUREG-1557, technical information and agreements on aging issues that resulted from the review of nine industry reports submitted by NUMARC in the early 1990's. The detrimental effects of aging and acceptable aging management programs are delineated in the text with the exception of 15 issues which have been high-lighted for continued analysis. Both the USNRC and industry have been using this report to support preparation and review of license renewal applications.

1. INTRODUCTION

In order to establish the United States Nuclear Regulatory Commission (USNRC) understanding of technical issues related to renewal of operating licenses for nuclear power plants, under Title 10, Part 54, of the U. S. Code of Federal Regulations (10 CFR Part 54) the USNRC reviewed and documented the detrimental effects of aging and the aging programs to manage these effects for certain systems, structures and components. The USNRC effort resulted in the publication of NUREG-1557, "Summary of Technical Information and Agreements from Nuclear Management and Resources Council Industry Reports Addressing License Renewal," in late 1996. This information on the detrimental effects of aging and the pertinent management programs were originally documented in the license renewal industry reports (IRs) submitted to the USNRC for review by the Nuclear Utilities and Resources Management Council (NUMARC) beginning in the late 1980's. The purpose of this paper is to summarize the information contained in NUREG-1557 and the notable findings and conclusions reached in the aforementioned document.

2. REPORT DEVELOPMENT

In the late 1980's, NUMARC, now the Nuclear Energy Institute (NEI), submitted for USNRC review ten IRs addressing aging issues associated with specific structures and components of nuclear power plants [1-10], and one IR addressing the screening methodology for performing an integrated plant assessment (IPA)[11], under Title 10, Part 54, of the

United States Code of Federal Regulations (10 CFR Part 54). The ten IRs for specific structures and components are:

1. Pressurized Water Reactor (PWR) Reactor vessel [1]
2. Boiling Water Reactor (BWR) Reactor Vessel [2]
3. PWR Containment [3]
4. BWR Containment [4]
5. PWR Reactor Coolant Pressure Boundary [5]
6. BWR Reactor Coolant Pressure Boundary [6]
7. PWR Reactor Vessel Internals [7]
8. BWR Reactor Vessel Internals [8]
9. Class I Structures [9]
10. Low-Voltage, In-Containment, Environmentally Qualified Cable [10]

The original intent of the IRs for specific structures and components was to serve as a referenceable surrogate for carrying out the integrated plant assessment (IPA) requirements of the USNRC license renewal rule, 10 CFR Part 54, as published in 1990. The IPA information to be submitted by a prospective applicant was to describe aging affects applicable to the systems, structures, and components within the scope of license renewal for their plant and to describe and justify the programs for managing those effects for a period of extended operation beyond the original 40 year operating license. The detrimental effects of aging on certain systems, structures, and components within nuclear power plants and the suggested programs for managing these detrimental effects of aging had been described in the IRs for use by individual utilities seeking to submit an application to the USNRC for a renewed operating license.

In 1992 the USNRC staff and industry resources were redirected and review of the IRs was terminated. USNRC and industry efforts were concentrated on revising the license renewal rule to focus on the effects of aging rather than an indeterminable number of aging mechanisms. Nonetheless, it was determined after finalization of the revised license renewal rule in 1995, that the effort already expended on review of the IRs should be utilized. The technical information and agreements reached at the point of review cessation, therefore, were to be incorporated into the draft USNRC standard review plan for license renewal (SRP-LR).

After a hiatus in IR review activities, NUREG-1557, "Summary of Technical Information and Agreements from Nuclear Management and Resources Council Industry Reports Addressing License Renewal," was completed. This report summarized the technical information and NUMARC/USNRC agreements reached from nine of the ten IRs, the cable IR [10] was excluded. The cable IR addresses the issue of environmental qualification (EQ) of electric equipment, which at the time of NUREG-1557 development, had been superseded by the USNRC EQ action plan to address aging of cables. The technical information and agreements documented in NUREG-1557 represent the status of the USNRC staff's review when the USNRC and industry resources were redirected to address license renewal rule implementation issues.

3. TECHNICAL INFORMATION AND AGREEMENT FORMAT

NUREG-1557 encompasses nine of the original ten NUMARC IRs, the relevant USNRC staff review documentation for each IR, and the NUMARC responses/positions taken with respect to the USNRC review. All the relevant documentation has been compiled and is listed in Appendix A of NUREG-1557. The technical information and agreements from the USNRC review have been compiled into tables, see examples Exhibits 1 and 2, and are presented in Appendix B of NUREG-1557. Each table consists of seven columns. The columns list among other things the aging-related degradation mechanisms (ARDMs) addressed in the IRs and their effects on structures and components. The effects of ARDMs were based primarily on information in the IRs. In summary the following "ARDMs" and their corresponding "effects" have been considered to affect structures and components in the Reactor Pressure Vessel (RPV), Reactor Vessel Internals, and the Primary Coolant Pressure Boundary (PCPB).

Aging Mechanism	Aging Effects
1. Corrosion, Microbiologically induced corrosion, Boric Acid[+] corrosion	Loss of material[++++]
2. Creep	Change in dimension
3. Erosion/Corrosion (E/C)	Wall thinning
4. Fatigue	Cumulative fatigue damage
5. Stress Corrosion Cracking (SCC)[++] (to include:IGSCC, TGSCC, & IASCC)	Crack initiation and growth
6. Neutron Irradiation Embrittlement[*]	Loss of fracture toughness
7. Stress Relaxation	Loss of preload
8. Wear[**]	Attrition
9. Thermal Embrittlement[(+++)(***)]	Loss of fracture toughness

[+]Boric acid wastage of external surfaces

[++]IGSCC-Intergranular SCC; TGSCC-Transgranular SCC; IASCC-Irradiation Assisted SCC.

[+++]Includes Cast Austenitic Stainless Steel (CASS) (BWR Primary Coolant Pressure Boundary)

[++++]Also listed as an Effect: "corrosion product buildup" (PWR Vessel Internals and BWR Pressure Vessel)

[*]Also listed as Irradiation Embrittlement (PWR Vessel Internals)

[**]Also listed as Fretting and Wear (PWR and BWR Pressure Vessel and BWR Vessel Internals) and Mechanical Wear (BWR and PWR Primary Coolant Pressure Boundary)

[***]Also listed as Thermal Aging (PWR Pressure Vessel)

The following "ARDMs" and their "effects" have been considered to affect structures and components in the Reactor Containment and Class I Structures:

Aging Mechanism	Aging Effects
Concrete Structures -	
1. Freeze Thaw	Scaling, cracking, and spalling
2. Leaching of Calcium Hydroxide	Increase of porosity & permeability
3. Aggressive Chemical Attack*	Increase of porosity & permeability, cracking, and spalling
4. Reaction with Aggregates	Expansion and cracking
5. Elevated Temperature	Loss of strength and modulus
6. Irradiation of Concrete	Loss of strength and modulus
7. Creep	Deformation
8. Shrinkage	Cracking
9. Corrosion	Loss of material
10. Abrasion and Cavitation	Loss of material
11. Restraint, Shrinkage, Creep, & Aggressive Environment	Cracking of masonry walls
12. Concrete Interaction with Aluminum	Loss of strength
13. Cathodic Protection Current	Cathodic protection effect on bond strength
Structural Steel & Stainless Steel Liner -	
1. Corrosion, Local corrosion, Atmospheric corrosion	Loss of material
2. Elevated Temperature	Loss of strength and modulus
3. Irradiation	Loss of fracture toughness
4. Stress Corrosion Cracking	Crack initiation and growth
Reinforcing Steel (Rebar) -	
1. Corrosion of Embedded Steel	Cracking, spalling, loss of bond, & Loss of material
2. Elevated Temperature	Loss of strength and modulus
3. Irradiation	Loss of strength and modulus
Miscellaneous -	
1. Fatigue	Cumulative fatigue damage
2. Settlement**	Cracking, distortion, increase in component stress level
3. Mechanical Wear	Lockup****
4. Strain Aging (of Carbon Steel)+	Loss of fracture toughness
5. Loss of Prestress***	Reduction of design margin
6. Corrosion of Steel Piles	Loss of material
7. Corrosion of Tendons	Loss of material

+Includes Stainless Steel-Bellows (PWR Containments)

*Also listed as "Aggressive Chemicals" (PWR Containments)

**Also listed as "Differential Settlement" (BWR Containments)

***Also listed as "Prestress Losses" (BWR Containments)

****Also listed as "Attrition" (PWR Containments)

Table B3. Brief summary of technical information and NUMARC/NRC agreements from PWR containment structures industry report

Aging-Related Degradation Mechanism	Aging Effects	Components	Materials[a]	NRC Comment Number[b]	NUMARC/NRC Agreement or Proposal	Basis for Agreement or Proposal
Aggressive Chemical Attack	Increase of porosity & permeability, cracking, & spalling	Concrete Containments Reinforced/Prestressed •Concrete Dome •Concrete Containment Wall Above Grade	Concrete	G-12, S-5, S-38 to S-41	For concrete containment structures that meet the basis requirements, aggressive chemical attack is non-significant ARDM.	Degradation caused by aggressive chemical attack is non-significant for concrete containment structures not exposed to aggressive environment (pH <5.5), or to chloride or sulfate solutions beyond defined limits (>500 ppm chloride,[5] and 1500 ppm sulfate);[6] or if exposed to ground water that exceeds the pH, chloride, sulfate limits the exposure is for intermittent periods only.
Aggressive Chemical Attack	Increase of porosity & permeability, cracking, & spalling	Concrete Containments Reinforced/Prestressed •Concrete Containment Wall Below Grade •Concrete Basemat Free-Standing Steel Containment with Flat Bottom & an Ice Condenser •Concrete Basemat	Concrete	G-10, G-13, G-15, S-25, S-36, S-37, S-41, S-65, S-66, S-69, S-72, S-75	Accessible concrete surfaces are periodically examined in accordance with the procedures of Type A[7] integrated leak rate test, or in accordance with ASME Sect. XI, Subsect. IWL.[8] Management for the effects of aggressive chemical attack of concrete surfaces that are not periodically examined due to inaccessibility requires further plant-specific evaluation.	In cases where containment concrete is exposed to aggressive groundwater (pH <5.5, chloride >500 ppm, & sulfate >1500 ppm), periodic inspection of accessible concrete surfaces as part of Type A integrated leak rate test performed under Appendix J, 10CFR50,[7] or in accordance with ASME Sect. XI, Subsect. IWL,[8] exam. category L-A, & guidelines of ACI 201.1.[9] Further evaluation for management of inaccessible areas is to be justified on a plant-specific basis.

EXHIBIT 1. Sample Table (B3) from NUREG-1557:PWR Containment Structures IR

Table B8. Brief summary of technical information and NUMARC/NRC agreements from BWR vessel internals industry report

Aging-Related Degradation Mechanism	Aging Effects	Components	Materials	NRC Comment Number[a]	NUMARC/NRC Agreement or Proposal	Basis for Agreement or Proposal
IGSCC	Crack initiation & growth	Access Hole Cover	Alloy 600	G-2, G-5, G-9, G-11, G-22, G-24,	GESIL 462S1[1] recommends volumetric inspections, implementation is plant specific, & recommended repair is to attach reinforcement hardware	Recommendations of GESIL 462S1[1] & safety analysis are current & effective inspection programs for detection & evaluation-repair of access hole covers.
		Core Shroud Head Bolts	SS, Alloy 600	G-27, G-28, G-29, G-33, G-34, S-3,	GESIL 433[2] recommends UT examination during outages, implementation is plant specific, & replacement is with crevice-free design.	Recommendations of GESIL 433[2] & replacement with crevice-free design are current & effective inspection programs for detection & evaluation-replacement of core shroud head bolts.
		Control Blade	SS	S-4, S-6, S-17, S-22, S-25,	GESIL 157 routine replacement,[3] operational parameter monitoring, inspection, evaluation, & replacement.[4]	Routine replacement & operational parameter monitoring are current & effective programs for detection & evaluation-replacement of control blades.
		Control Rod Drive (CRD) Housing	SS	S-26 to S-29, S-31, S-32, S-38,	ASME Sect. XI[5] requires volumetric exam. of welds & VT-2 of pressure retaining boundary & system leakage & hydro-static tests.	ASME Sect. XI, Subsect. IWB,[5] exam. categories B-O & B-P is current & effective program for detection & evaluation-repair-replacement of CRD housing.
		Core Spray Sparger	SS	S-49, S 54, S 55	NRC Bulletin 80-13[6] recommends visual inspection during refueling outages; analytical evaluation; & repair.	NRC Bulletin 80-13[6] & safety analysis are effective inspection programs for detection, evaluation-repair of core spray sparger.
		Intermediate Range Monitor/ Source Range Monitor (IRM/SRM) Dry Tubes	SS		GESIL 409, Rev. 1[7] recommends visual inspection; leakage monitoring; replacement is with crevice-free design; & resistant material.	Recommendations of GESIL 409,[7] leakage monitoring, & replacement with crevice-free design, are effective inspection programs for detection & evaluation-replacement of dry tubes.

EXHIBIT 2. Sample Table (B8) from NUREG-1557:BWR Vessel Internals IR

In addition, the tables list the specific structures and/or components, and their construction materials. For each IR, a complete list of what are defined as structures and components is included in the end of the table (see Exhibit 3). In general, the IRs present only representative examples and do not provide a comprehensive list of the type, grade, and specification of materials used for various reactor structures and components but give a good baseline from which comparative reviews by a prospective license renewal applicant can begin. For most IRs, only material categories such as stainless steel (SS), cast austenitic stainless steel (CASS), Ni Alloy, or carbon steel (CS), are described. A detailed list of material type and grade is, however, provided in the PWR and BWR reactor vessel IRs.

LIST OF BWR CONTAINMENT COMPONENTS:

MARK I STEEL CONTAINMENT
Drywell Interior Surface
Drywell Exterior Surface
Drywell Head
Embedded Shell Region
Drywell Support Skirt
Sand Pocket Region
Torus Interior Surface
Torus Interior Surface at Waterline
Torus Exterior Surface
Torus Ring Girder
Vent Lines
Vent Line Bellows
Vent Header
Downcomers and Bracing
Vent System Supports
Torus Seismic Restraints
Torus Support Columns/Saddles
ECCS Suction Header
Ocean Plant with Uncoated CS Surfaces
Uncoated Submerged CS Surfaces

MARK II STEEL CONTAINMENTS
Drywell Interior Surface
Drywell Exterior Surface
Drywell Head
Suppr. Chamber Exterior Surface
Suppr. Chamber Interior Surface
Suppr. Chamber Interior Surface at Waterline
Region Shielded by Diaphragm Floor
Embedded Shell Region
Sand Pocket Region
Support Skirt
Downcomer Pipes and Bracing
Ocean Plant with Uncoated CS Surfaces
Uncoated Submerged CS Surfaces

MARK I CONCRETE CONTAINMENT
Drywell Liner Interior Surface
Drywell Liner Exterior Surface
Torus Liner Interior Surface
Torus Liner Interior Surface at Waterline
Torus Liner Exterior Surface
Liner Anchors
Drywell Concrete
Torus Concrete
Drywell Concrete Reinforcing Steel
Torus Concrete Reinforcing Steel
Vent Lines
Vent Line Bellows
Vent Headers
Downcomers and Bracing
Vent System Supports
Drywell Head

MARK II CONCRETE CONTAINMENTS
Drywell Liner Interior Surface
Drywell Liner Exterior Surface
Suppr. Chamber Liner Interior Surface
Suppr. Chamber Liner Interior Surface at Waterline
Suppr. Chamber Liner Exterior Surface
Liner Anchors
Liner Region Shielded by Diaphragm Floor
Containment Concrete
Concrete Containment Reinforcing Steel
Drywell Head
Downcomer Pipes and Bracing
Concrete Basemat
Basemat Liner
Basemat Reinforcing Steel
Prestressing Tendons and Ducts

MARK III STEEL CONTAINMENTS
Containment Shell Interior Surface
Containment Shell Exterior Surface
Suppr. Chamber Shell Interior Surface
Suppr. Chamber Shell Exterior Surface
Basemat Liner
Liner Anchors
Concrete Basemat
Concrete Fill in Annulus
Embedded Shell Region

MARK III CONCRETE CONTAINMENTS
Containment Liner Interior Surface
Containment Liner Exterior Surface
Suppr. Chamber Liner or Cladding Interior Surface
Suppr. Chamber Liner Exterior Surface
Concrete Containment Wall Above Grade
Concrete Containment Wall Below Grade
Concrete Dome
Basemat Liner
Concrete Basemat
Liner Anchors
Containment Wall Reinforcing Steel
Dome Reinforcing Steel
Basemat Reinforcing Steel

COMMON COMPONENTS
Penetration Sleeves
Dissimilar Metal Welds
Penetration Bellows
Personnel Airlock
Equipment Hatches
CRD Hatch

EXHIBIT 3. Sample Listing of IR Components: BWR Containment Components

The USNRC staff review of each original IR resulted in a set of comments/issues for each IR which are referenced in NUREG-1557 in order to recreate the lineage of the aging issue agreement. NUMARC/USNRC agreements or proposals on whether a ARDM or ARDM/component combination is potentially significant, and if it is potentially significant, is given and a brief description of the program that can adequately manage the effects of aging is presented in a similar fashion. The technical basis for these agreements or proposals, including assumptions and references, are also described in brief in NUREG-1557. A few examples of the information delineated in the report as aging management programs and their bases are given below.

1. For a specific ARDM or ARDM/component combination, if the effects of aging are not potentially significant, "non-significant" is listed in the agreements column. The technical basis, assumptions, and references for the agreement are presented in column seven. For example, the effect of creep is non-significant for BWR primary coolant piping and fittings fabricated from carbon steel (CS) or stainless steel (SS) because the reactor operating temperatures are significantly lower than the temperatures at which creep is a concern to CS and for SS components. Also, if the effects of aging are not potentially significant when certain bounding conditions are met, then "for components that meet the basis requirements, this ARDM is non-significant" is listed in the agreements column. For example, the effects of freeze thaw is non-significant for Class I concrete structures that meet the following criteria: located in geographic regions of negligible weathering conditions (weathering index < 100 day-inch/year): and if located in severe weathering conditions (weathering index 100-500 day-inch/year) the concrete mix design meets the air content and water-to-cement ration requirements of American Concrete Institute (ACI) 318-63 or ACI-349-85.

2. If a specific ARDM/component combination is potentially significant and the effects of aging are adequately addressed by current management programs, then a brief description of the program is provided in column six. For example, the program delineated in NUREG-0313, "Technical Report on Material Selection and Processing Guidelines for BWR Coolant Pressure Boundary Piping," July 31, 1977, and implemented through USNRC Generic Letter 88-01, "USNRC Position on IGSCC in BWR Austenitic Stainless Steel Piping," January 25, 1988, is a current and adequate program to manage the effects of intergranular stress corrosion cracking (IGSCC) of SS piping and fittings of BWR primary coolant pressure boundary.

3. If a specific ARDM/component combination is potentially significant and the current programs are not adequate for managing the effects of aging, then column six simply states "current practices to be enhanced, select plant-specific aging management." For these cases, the NUMARC recommended aging management options are described in Chapter 6 of the IRs. However, chapter six was not the focus of the USNRC review of the IRs.

4. An ARDM or ARDM/component combination is listed as "unresolved issue" if no agreement was reached between NUMARC and the USNRC staff. For these cases, both the NUMARC and USNRC proposals are briefly described in the agreements column. An example of an unresolved issue is the effects of thermal aging embrittlement on PWR primary coolant system components fabricated from cast austenitic stainless steel (CASS). The NUMARC proposal considers a ferrite content screening criterion and American Society of Mechanical Engineers (ASME) Code Section XI, Subsection IWB, inspection to be an adequate program for managing the effects of thermal embrittlement. The USNRC proposal, however, considers that ferrite content criterion is inadequate for screening and VT-3 visual examination is not intended or reliable for detecting tight cracks.

4. SUMMARY AND OBSERVATIONS

Nine of the ten IRs submitted by NUMARC addressing the detrimental effects of aging associated with specific structures and components of nuclear power plants were reviewed by the USNRC. The technical information and NUMARC/USNRC agreements for each IR have been compiled into tables (Appendix B to NUREG-1557). The information presented in each of the tables includes specific structures and components and their materials of construction; ARDMs and their effects on structures and components; relevant comments of the USNRC staff; and the NUMARC/USNRC agreements or proposals and their technical bases, including assumptions and references.

Considerable effort was expended by industry representatives and the USNRC staff to come to agreement on the aging issues pertinent to several major structures and components in nuclear power plants. These aging issues and associated structures and components are of prime importance for a prospective applicant aiming to satisfy the requirements of the revised license renewal rule, 10 CFR Part 54, published in 1995. NUREG-1557 documents the USNRC understanding of the information delineated in the IRs and the agreed upon and disagreed upon technical positions from the previous USNRC review effort. Resulting from the industry/USNRC exchanges it was determined that fifteen open technical issues existed. These open technical issues include the following:

(1) Fatigue in Metal Components
(2) Environmental Qualification of Cables
(3) Thermal Aging Embrittlement of Cast Austenitic Stainless Steel Components
(4) Irradiation-Assisted Stress Corrosion Cracking of Reactor Internals Components
(5) Stress Relaxation of PWR Internals Components
(6) Primary Water Stress Corrosion Cracking of High-Nickel Alloys
(7) Stress Corrosion Cracking in PWR Metal Components
(8) Neutron Irradiation Embrittlement (Definition of Reactor Vessel Beltline)
(9) Freeze-Thaw Damage in Concrete Structures (Significance of Effects)
(10) Alkali-Aggregate Reactions in Concrete Structures (Significance of Effects)
(11) Differential Settlement of PWR Containments and Class I Structures (Significance of Effects)
(12) Reinforcement Corrosion in PWR Concrete Containments (Significance of Effects)
(13) One Time Inspections of Concrete and Steel Structures
(14) Ultrasonic Inspection of Pressure Vessels and Components
(15) Visual Inspection of Components and Structures using certain American Society of Mechanical Engineers Code Acceptance Criteria

Resolution of these technical issues have been slated to be the focus of USNRC license renewal reviews. Resolution of these technical issues, for a prospective license renewal applicant, is paramount to successfully meeting the requirements of the revised license renewal rule, 10 CFR Part 54. Several of these issues are also currently under review by the USNRC for applicability to operating reactors for their existing 40 year operating license.

In the future, as U.S. utilities begin preparation of an application for license renewal, NUREG-1557 will have established a starting point to begin an aging management analysis. Documented in NUREG-1557 are the aging management programs deemed acceptable by the USNRC and the technical issues where the USNRC and the industry hold diverse positions. It is resolution of the open technical issues elucidated by this report that is necessary to provide a stable and predicable regulatory environment for a license renewal applicant. By referencing this publicly available report, a prospective license renewal applicant can become familiar with the level of effort necessary to initiate aging studies of other systems, structures, and components that may also be subject to the requirements of the license renewal rule. NUREG-1557 can be used as a time saving and a cost cutting mechanism by a license renewal applicant. In addition to the benefits for industry utilities, the USNRC is currently considering incorporation of the appropriate technical information and agreements as the basis for a revised draft USNRC SRP-LR. This new SRP-LR will establish the methods and acceptance criteria that meet the requirements of 10 CFR Part 54 that the USNRC staff will utilize to perform a review of future license renewal applications.

5. CONCLUSIONS

Industry submittal and subsequent USNRC review of nine of the ten IRs resulted in the establishment of a clear understanding of industry and USNRC positions related to the detrimental effects of aging and the programs necessary for managing them. NUREG-1557 concisely summarizes these agreements and disagreements in a format usable to a prospective license renewal applicant. The information delineated may afford a prospective applicant an introduction to the significant aging issues needing resolution. Although several aging issues remain unresolved the report contains valuable information that may be referenced by a prospective applicant saving them both time and resources. It should also be noted that the USNRC will consider utilizing this report by incorporating resolved aging issues into a draft SRP-LR that will meet the requirement of the revised license renewal rule, 10 CFR Part 54.

6. REFERENCES

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3. Nuclear Management and Resources Council. September 1991. Pressurized Water Reactor Containment Structures License Renewal Industry Report, August 1989. Revision 1. Report Number 90-01.

4. Nuclear Management and Resources Council. December 1991. Boiling Water Reactor Containments License Renewal Industry Report, July 1990. Revision 1. Report Number 90-10.

5. Nuclear Management and Resources Council. May 1992. PWR Reactor Coolant System License Renewal Industry Report, October 1990. Revision 1. Report Number 90-07.

6. Nuclear Management and Resources Council. April 1992. BWR Primary Coolant Pressure Boundary License Renewal Industry Report, September 1990. Revision 1. Report Number 90-09.

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