APPENDIX D

DISPOSITION OF FIVE UNION OF CONCERNED SCIENTISTS REPORTS

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D.1. Introduction

In a letter dated May 5, 2000, the Union of Concerned Scientists (UCS) provided five reports (ADAMs accession number ML003713188) to be considered for the development of the improved license renewal guidance documents. The titles of these documents are included in Section D.3 of this appendix. The components and aging effects provided in these reports were evaluated, and the results of this review are summarized in this appendix.

D.2. EVALUATION AND DISPOSITION OF COMMENTS

Table D, at end of Appendix D, contain the evaluation and disposition for each of the UCS reports. The column heading “Document Number” is primarily intended to provide the source of the comment, meaning the report being reviewed; it provides a means of referring to each report without having to use the title. For example, UCS-1 indicates that the report being reviewed is from UCS, and the “1” segregates this report from all other UCS reports. The references in Appendix D.3 provide the sources of all comments

D.3 REFERENCES

The following references were included in the Union of Concerned Scientist's letter (ADAMs accession number ML003713188):

1. H. M. Thomas, Rolls-Royce & Associates, "Pipe and Vessel Failure Probability," Reliability Engineering, 1981.

2. Nicholas T. Saltos, Probabilistic Safety Assessment Branch, Nuclear Regulatory Commission, "Risk Impact of Environmental Qualification Requirements for Electrical Equipment at Operating Nuclear Power Plants," March 30, 1993.

3. Robert Pollard, Union of Concerned Scientists, "US Nuclear Plants —- Showing Their Age / Case Study: Core Shroud Cracking," September 1995.

4. Robert Pollard, Union of Concerned Scientists, "US Nuclear Plants —Showing Their Age / Case Study: Reactor Pressure Vessel Embrittlement," December 1995.

5. Robert Pollard, Union of Concerned Scientists, "US Nuclear Plants —Showing Their Age / Case Study: Steam Generator Corrosion," December 1995.

Table D: Disposition of Five Union of Concerned Scientists Reports

Document No.	Item Number	Document Title	Document Summary	NRC Disposition
UCS-1	IV.C1.1.1-IV.C1.1.11, IV.C1.1.13, IV.C2.2.1-IV.C2.2.8, IV.D1.1.5, IV.D2.1.5, V.D2.1.1-V.D2.1.7, VII.E2.1.1 VII.E3.1.1.	H. M. Thomas, Rolls-Royce & Associates, “Pipe and Vessel Failure Probability,” Reliability Engineering, 1981.	This document presents a generalized approach to estimation of failure probabilities for leakage and ruptures of piping and vessels. Failure data includes stress corrosion cracking of boiling water reactor (BWR) piping and fatigue cracking of light water reactor (LWR) piping. Steam generator tube failures are also discussed in the paper.	Most of the failure data presented in this document are associated with failures in the first few years of life resulting from design and fabrication defects, thus are not aging management issues. Most pressure vessel failures reported in this document were due to manufacturing defects, not to any aging effects, and they had occurred in fossil power plants (Reference 2 of the document: WASH 1318, *Technical report on analysis of pressure vessel statistics from fossil-fuelled power plant service and assessment of reactor vessel reliability in nuclear power plant service*, USAEC Report, 1974.) Steam generator tube failures are mentioned in the document without identifying the associated aging mechanisms. For these reasons, the role of aging degradation in the reactor pressure vessel failures and steam generator tube failures discussed in this document cannot be evaluated. The GALL report contains comprehensive evaluation of the existing aging management programs for both reactor pressure vessels and steam generator tubes discussed in this document. The GALL report also contains comprehensive evaluation of aging management programs for SCC of BWR piping and fatigue and corrosion of LWR piping. The GALL report has not been revised to address the review of this document.

Table D: Disposition of Five Union of Concerned Scientists Reports (continued)

Document No.	Item Number	Document Title	Document Summary	NRC Disposition
UCS-2	IV.A1.2.7, IV.A1.5.5, IV.B1.1.1, IV.B1.1.2, IV.B1.1.4, IV.B1.2, IV.B1.3.1 IV.B1.3.2, IV.B1.4.1- IV.B1.4.9, IV.B1.5.1, IV.B1.5.2, IV.B1.6.1- IV.B1.6.4.	*Robert Pollard, Union of Concerned Scientists, "US Nuclear Plants – Showing Their Age / Case Study: Core Shroud Cracking," September 1995.*	This document focuses on aging of BWR vessel internals: steam dryer, steam separator and its support ring, core shroud, shroud head, core plate, top guide, feedwater sparger, core spray line and sparger, jet pump assemblies including jet pump sensing line, fuel supports, incore neutron flux monitors (housings, dry tubes, and guide tubes), neutron source holder, control blade, and CRD housing. The document listed the following aging effects and mechanisms for the internals components: crack initiation and growth due to SCC and fatigue, loss of fracture toughness due to neutron irradiation and thermal aging embrittlement, loss of material due to erosion, and deformation due to thermal creep.	Most of the internals and aging mechanisms addressed in this document are included in GALL Chapter IV B1, but some are not. Six of the internals mentioned in this document (steam dryer, steam separator and its support ring, steam shroud head and bolts, and feedwater sparger) are not included in GALL because they have no license renewal intended function (not safety related and not a part of the pressure boundary) The correct name for steam separator support ring is holddown beams, which are attached to the vessel top head. These attachment welds are included in Chapter IV-A1 of GALL. Control blades are not included because they are short-lived components and are replaced periodically during plant operation. Neutron source holders are not included because most BWR plants have removed them from the vessels. Creep of BWR internals is not included because the temperatures experienced by the internals are well below the temperature at which creep is a concern for stainless-steel components. Erosion of jet pump assemblies is not included because there has been no evidence of erosion in the jet pump throat area, which is the most susceptible location for erosion. Even if erosion occurs in the throat area, it will not impair the intended function of the jet pump, which is to reflood the core to two-thirds core height during an accident. SCC of fuel support pieces is not included because they are made of cast austenitic

Table D: Disposition of Five Union of Concerned Scientists Reports (continued)

Document No.	Item Number	Document Title	Document Summary	NRC Disposition
UCS-2 (cont.)				stainless steel and/or subjected to low stresses. The GALL report was modified to address the review of this document by including the incore neutron flux monitor guide tubes and a jet pump sensing line.
UCS-3	IV.A2.5.1, IV.A2.5.2.	Robert Pollard, Union of Concerned Scientists, “US Nuclear Plants — Showing Their Age / Case Study: Reactor Pressure Vessel Embrittlement,” December 1995.	This document reviews information pertaining to reactor pressure vessel embrittlement and the issues related to the safe operation of nuclear power plants.	Aging management of neutron embrittlement of PWR and BWR reactor pressure vessels has been addressed, respectively, in GALL, Chapters IV-A1 and IV-A2. The GALL report was not revised to address the review of this document.
UCS-4	IV.D1.2.1, IV.D1.2.3, IV.D2.2.1, IV.D2.2.2.	Robert Pollard, Union of Concerned Scientists, “US Nuclear Plants — Showing Their Age / Case Study: Steam Generator Corrosion,” December 1995.	This document reviews aging degradation of PWR recirculating steam generator tubes. The document mentions that the tubes in once-through steam generators have experienced similar types of aging degradation but does not provide any specific information. The document identifies two issues related to aging management of steam generator tubes: (1) Quality of current inspection techniques for detecting steam	All but one degradation mechanisms for steam generator tubes were included in GALL; for recirculating steam generator tubes in Chapter IV D1 and for once-through steam generator tubes in Chapter IV D2. Loss of section thickness due to fretting (wear) of once-through steam generator tubes is now included in Chapter IV D2 because fretting has caused material loss in these tubes and challenged their structural integrity. Regarding the quality of current inspection techniques for detecting steam generator tube degradation, the GALL report has been revised to recommend further evaluation of the effectiveness of the proposed aging management programs during license renewal period for steam generator tubes.

Table D: Disposition of Five Union of Concerned Scientists Reports (continued)

Document No.	Item Number	Document Title	Document Summary	NRC Disposition
USC-4 (cont.)			generator tube degradation, (2) Quality of current inspection techniques for detecting steam generator tube degradation, (3) Adequacy of the alternate repair criterion based on voltage rather than crack size.	The second issue mainly applies to the specific case of ODSCC in Westinghouse drill-hole support plates. The alternate Repair criteria were developed only after a substantial database had been developed to demonstrate that using such a criterion maintained the margin of 3 delta p against burst that has always been required for SG tubing and that leakage could be kept low enough to ensure that radiation exposure limits to the public are not violated. This issue does not warrant any additional changes in GALL than the one mentioned above. The GALL report has been revised to address the review of this document.
UCS-5	IV.C1.1.13, IV.C2.1.5, IV.C2.2.8.	Nicholas T. Saltos, Probabilistic Safety Assessment Branch, Nuclear Regulatory Commission, “Risk Impact of Environmental Qualification Requirements for Electrical Equipment at Operating Nuclear Power Plants,” March 30, 1993.	This document used probabilistic risk assessment (PRA) techniques to quantify the risk impact of electrical equipment qualified under the “old” EQ requirements and compare to recent requirements. The document also identified equipment in the containment whose failure could impact risk important operations.	Review of this document has resulted in addressing aging of instrumentation lines in GALL. These lines are included in GALL as small-bore piping in Chapter IV. There has been a clarification of the treatment of small bore piping and instrument lines in Chapters V, VII, and VIII of the GALL report. The GALL report has been revised to address the review of this document.

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