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UNITED STATES OF AMERICA
NUCLEAR REGUALTORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In The Matter Of	)	
	)	
	)	
COMMONWEALTH EDISON COMPANY	)	Docket Nos. 50-454 OL
	)	50-455 OL
	)	
(Byron Nuclear Power Station,	)	
Units 1 & 2)	)	
	)	

AFFIDAVIT OF DANIEL D. MALINOWSKI

The attached answers to the questions posed by counsel constitute my testimony in the above-captioned proceeding. The testimony is true and accurate to the best of my knowledge, information and belief.

Daniel D Malinowski

Daniel D. Malinowski

Subscribed and sworn to
before me this 19 day
of July, 1982.

PAULETTE SLOWIKA, NOTARY PUBLIC
MONROEVILLE BORO, ALLEGHENY COUNTY
MY COMMISSION EXPIRES MARCH 10, 1986
Member, Pennsylvania Association of Notaries

AFFIDAVIT OF DANIEL D. MALINOWSKI

CONCERNING DAARE/SAFE

CONTENTION 9c.

Q.1. State your name and address.

A.1. My name is Daniel D. Malinowski. My business address is P.O. Box 855, Pittsburgh, Pennsylvania 15230. I am employed by Westinghouse Electric Corporation as a Manager of Field Data Analysis in the Steam Generator Systems and Materials Engineering Department of the Nuclear Technology Division.

Q.2. State your educational background and professional work experience.

A.2. I was graduated from Duquesne University in 1959 with a B.S. Degree in Chemistry. In 1963 I received an M.S. in Chemistry from the Massachusetts Institute of Technology, where my area of specialization was Nuclear Chemistry.

From February 1963 until March 1967, I was employed by Westinghouse Electric Corporation at the Bettis Atomic Power Laboratory. I worked as a Scientist in the Radiochemistry section where my activities included radiochemical assay, computer analysis of plant follow data, and supervision of radiochemical evaluation of naval reactor cores at initial criticality testing.

From March 1967 to June 1968, I was employed by the Tracerlab Division of Laboratory for Electronics (LFE, Inc.) in Waltham, Massachusetts as a Staff Chemist in the Technical Services Department. My duties included supervision of Radiochemical Laboratory activities, development of radioassay procedures, and direction of subcontract work in radiochemical evaluation of naval nuclear cores.

From July 1968 to December 1973 I was employed as a Senior Engineer by Westinghouse Nuclear Energy Systems in the Chemistry and Chemistry Operations groups at Forest Hills, Pennsylvania.

From December 1973 to October 1975, I worked on analyses of steam generator ("SG") in-service experience and inspection trends evaluation. In October 1975, after my appointment as a Principal Engineer, I organized and directed a task force whose goals were to assemble and evaluate design, inspection, and operational data from Westinghouse-designed SG's.

Q.3. Please describe your work responsibilities as they concern steam generator tube integrity matters in more detail.

A.3. Since December 1973, I have been employed in steam generator related programs; I have assisted in the overall coordination of laboratory and in-plant activities relating to corrosion studies, steam generator inspections, and chemistry specifications, and I have participated in assembly and evaluation of operating plant data from steam generators. My duties have included presentation of Westinghouse chemistry recommendations to present and future operators, supervising development of an evaluation of nondestructive testing methods for SG tubes, especially eddy current testing, and expediting automated methods of data evaluation, processing and transmission. In June 1977, a SG Data Analysis Group was formed -- with me as manager as part of the Nuclear Steam Generation Division. This activity encompassed diagnostic reevaluation of eddy current inspection tapes, development of improved eddy current techniques, and evaluation of inspection results from operating plants.

Q.4. What is the purpose of your affidavit?

A.4. My affidavit addresses DAARE/SAFE Contention 9c. insofar as that contention concerns the various phenomena that affect steam generator tube integrity and the water chemistry measures that can be taken either to eliminate or to minimize the adverse effects of these phenomena. My testimony supplements the testimony of Mr. Emmett L. Murphy of the NRC Staff. Mr. Murphy's testimony was submitted as part of a joint affidavit in support of the Staff's June 7 motion for summary disposition.

Q.5. How many operating nuclear power reactors are equipped with Westinghouse steam generators?

A.5. There are approximately 76 PWR plants of Westinghouse design in operation containing some 224 steam generators as integral parts of the NSSS for those plants.

Q.6. Are all these steam generators of the same design?

A.6. No. Broadly speaking, there are two basic types of SG's in service, the feedring and the preheater types. The Byron Stations will employ SG's of the preheater type, models D4 and D5. Model D4's are installed in Unit 1 and D5's in Unit 2.

Q.7. Please explain the design difference between model D4 and D5 SG's.

A.7. Materials in the D4 SG's are the same as most of the earlier SG's, consisting of Inconel 600 tubing with carbon steel support plates. The D4 SG's represent a stage in the evolution of the steam generator design which incorporates improved thermal characteristics. The tubes are full depth expanded in the tube sheet and the blowdown internal piping has been relocated. The relocation of blowdown pipes to facilitate removal of sludge from the SG is expanded to diminish the potential for tube corrosion. Higher efficiency is achieved by incorporation of an integral preheater.

The D5 model preheat SG is also a design evolution from the D4 which incorporates recent design improvements. For example, the support plates in the D5 are manufactured from Type 405 stainless steel, a more corrosion-resistant material than the carbon steel used in the D4 model. The tubing material, Inconel 600 for both models as been thermally treated in the D5 to provide additional resistance to stress corrosion cracking. The design of the support plates has been altered from the cylindrical hole/flow hole pattern in D4's to the Quatrefoil hole wherein the bulk liquid flow passes directly over the tube surfaces to reduce the potential for local concentration of chemicals.

Q.8. What are the phenomena that may cause steam generator tube degradation?

A.8. Generally five such degradation phenomena have been observed in operating plant SG's. These are:

1. intergranular corrosion
2. thinning or wastage
3. pitting
4. denting
5. wear

Q.9. Starting with intergranular corrosion, please describe the nature and characteristics of this phenomenon and any corrective means that have been taken to mitigate or prevent its occurrence.

A.9. Intergranular corrosion occurs either in the form of stress corrosion cracking (SCC) or as intergranular attach (IGA). SCC may be described as a linear separation of the metal grains which is strongly dependent on the environment and stresses in the region of the crack. IGA usually affects an extended area of the tubing and is manifested as grain boundary dissolution, leaving the tube largely in its original dimensions but partly lacking in structural integrity.

On the outside diameter ("OD") surfaces, IGA and SCC occur in regions where highly alkaline environments persist. These two phenomena often occur together, though several cases are known where OD SCC has occurred alone. Measures which cleanse the affected areas -- (tubesheet crevices and sludge pile regions in a SG) have been helpful in ameliorating SCC; these include sludge lancing, water fill and drain sequences and hot and cold water soaks. The continuation of IGA has been diminished by the combined application of crevice flushing techniques and reduced temperature operation.

The occurrence of SCC on the inside diameter

("ID") surfaces of the tubes appears to occur in locations where the tubing has experienced some cold work or where extensive denting may produce plastic deformation or ovalization of the tubing. Measures which reduce residual stresses in transition and target zones, such as thermal treatment and minimizing cold work, are effective in limiting ID cracking.

Q.10. Please continue your answer with respect to thinning or wastage.

A.10. Localized thinning of the tubing from the OD surface was largely associated with the use of sodium phosphate for secondary system chemistry control. With the advent of All Volatile Treatment (AVT) for water chemistry control, the incidence of tube wall thinning has diminished.

A few plants using AVT in recent years have developed instances of wall thinning similar to phosphate wastage. These observations were in the support plate intersections as compared with the predominance of thinning in sludge pile regions observed when phosphates were used. The tubes affected are confined to the peripheral region of the bundle and the corrosion rate experienced has not been excessive. This phenomenon is being closely monitored and efforts are in progress at several stations to understand and control it.

Q.11. Please continue your answer with respect to "pitting."

A.11. Pitting of the tube surface has been observed in only one Westinghouse plant. It is believed to be the result of high dissolved oxygen in an acidic chloride environment. Control of pitting can be accomplished by rigorous adherence to water chemistry guidelines to eliminate air ingress and condenser inleakage.

Q.12. Please continue your answer with respect to "denting."

A.12. Denting of SG tubes results from corrosion of the carbon steel support plates around the tubes. The magnetite (Fe_3O_4) thus produced in an acid chloride medium fills the annulus between the tube and the support plate. The linear corrosion process continues to generate Fe_3O_4, creating pressure on the tube and resulting in localized plastic deformation, i.e., denting. If the localized strains become excessive, tube leakage may result from ID-SCC.

Denting is controlled by elimination of condenser inleakage -- the source of acid chlorides, and air inleakage. In some cases improvements in SG chemistry have been achieved by using full flow condensate polishing to reduce inleakage contaminants.

Q.13. Please continue your answer with respect to "wear."

A.13. Wear phenomena can be caused by flow induced vibration, impingement of loose parts and foreign objects. Tube wear caused by flow induced vibration has been observed at antivibration bar (AVB) locations in the upper tube bundle. This can be controlled by tightening the upper bundle tube array; in one case this was accomplished by insertion of additional sets of AVB's. Impingement of loose parts and foreign objects on the tube surface is effectively eliminated by control of maintenance operations and through rigorous administrative procedures which require full accounting for all materials, parts, and tools which are introduced into a SG. A visual-photographic-television examination of potential regions in the SG where such objects might fall may be considered following maintenance operations.

Tube wear caused by flow induced vibration has been observed in model D SG's. This phenomenon is undergoing review and it is discussed in Mr. Edward Burns' Testimony.

Q.14. How important are water chemistry measures in controlling steam generator tube degradation?

A.14. Water chemistry control is extremely important. As I have indicated earlier in my testimony, control of the various potential forms of corrosion described above is achieved through rigorous application of the AVT chemistry guidelines; this includes commitments by the utility to minimize air inleakage and condenser inleakage to prevent excessive ingress of corrodents into the steam generators. Routine laboratory surveillance of the key chemistry parameters as well as automatic instrumentation to provide rapid and continuous on-line readout will permit timely responses to any chemical transients as they occur. Prompt corrective actions to eliminate the source of contaminant ingress are required to minimize the impact of such events on tube integrity.