July 20, 1981

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of	§	
	§	
HOUSTON LIGHTING & POWER COMPANY	§	Docket No. 50-466
	§	
(Allens Creek Nuclear Generating Station, Unit 1)	§	
	§	

DIRECT TESTIMONY OF REY L. CALL
ON DOHERTY CONTENTION 26 - STUD BOLTS

Q. Please state your name and place of employment.

A. My name is Rey L. Call and I am employed by the General Electric Company as Principal Engineer in the Availability Engineering Subsection. Also, it is important to note that I was the Manager of GE's Reactor Pressure Vessel Design unit at the time the Allens Creek RPV was being fabricated.

Q. Would you describe your professional qualifications?

A. A copy of my professional qualifications are set forth as Exhibit RLC-1 to this testimony.

Q. What is the purpose of your testimony?

A. The purpose of my testimony is to address Mr. Doherty's Contention No. 26 which asserts that the reactor vessel and stud bolts are inadequate in the following areas:

a) Yield strength is exceeded during anticipated

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transient without scram (ATWS)

b) Primary tensile component of the total bolt stress has been inadequately considered.

c) The total stress in the bolt as well as the strain energy in the reactor vessel head flange are such that the failure of one stud bolt will cause the failure of all the stud bolts.

Q. Have any stud bolts ever failed in a BWR?

A. The only instance of stud bolts failure occurred at the La Crosse BWR. That failure was attributable to insufficient fracture toughness resulting from faulty heat treatment in combination with a corrosive attack on the bolts caused by the breakdown of the silver plating. These problems have been addressed by the NRC and resulted in Regulatory Guide 1.65. Allens Creek complies with Reg. Guide 1.65 as stated in Appendix C of the PSAR.

Q. What criteria govern the design of the Reactor Pressure Vessel (RPV) stud bolts?

A. The design of the RPV stud bolts is governed by Title 10 of the Code of Federal Regulations, Part 50.55a and Appendix A, Criterion 30, which requires the use of the ASME Boiler and Pressure Vessel Code, Division I, Section III and Section XI. In addition Regulatory Guide 1.65 issued October 1973 supplements the requirements of the ASME Code,

Section III and Section XI.

Q. Of what material are the bolts made?

A. The bolts are a special quality Chromium-Nickel-Molybdenum alloy steel which has been specified for use in nuclear applications by the ASME Code. The ASME Code specification for this material is SA540, Class 3, Grade B23 or B24. This specification places requirements on mechanical properties, chemical content, heat treatment, and on many other aspects connected with the manufacture of the material. Regulatory Guide 1.65 supplements this ASME specification.

Q. What mechanical property requirements of the ASME Code and Regulatory Guide 1.65 are pertinent to this contention?

A. There are basically four material properties of importance: (1) yield strength, which is a measure of the stress at which permanent strain deformation will occur; (2) tensile strength which is a measure of the maximum nominal stress a specimen supports in a tension test prior to fracture; (3) fracture toughness, which is a measure of the material's resistance to the extension of a crack; and (4) hardness, which is a measure of the material's resistance to penetration.

Q. What tests are performed to guarantee that the stud bolts will have adequate material properties?

A. For tensile and fracture toughness testing one

test is made for each "lot" of material. A lot is one batch of material heat treated in the same charge or as one continuous operation. Hardness tests are performed on each stud after heat treatment but before final machining. A chemical analysis of the stud material is also done to check that the chemical composition of the material meets the material specification requirements.

Q. Besides mechanical properties, what other properties or processes are considered in fabricating stud bolts in order to prevent failure in service?

A. In order to resist corrosion, each stud bolt has a maganese phosphate coating which is applied as a measure to prevent oxidation. Regulatory Guide 1.65 specifically endorses this material for use as a coating. The thread lubricant, which is a colloidal suspension of graphite in isopropyl alcohol, is also in compliance with Regulatory Guide 1.65.

To provide resistance to stress corrosion cracking, the ACNGS stud bolts are tempered to a maximum tensile strength of 170 KSI. The 170 KSI tensile limit is included in the bolt design and ensured by prescribed testing.

The bolts will undergo fabrication inspections, pre-service inspections, and inservice inspections. The fabrication inspection is done in accordance with ASME

Section III, Subarticle NB-2580, as supplemented by Regulatory Guide 1.65. Included are a visual examination, and either a magnetic particle test (MT) or a liquid penetrant test (PT), after final heat treatment. The pre-service inspection is done in accordance with ASME Section XI, which requires a MT surface examination and a volumetric, ultrasonic test (UT) examination. The pre-service inspection serves as a baseline for future inservice inspection. The inservice inspections are also governed by ASME Section XI and involve MT and UT examination.

Finally the reactor pressure vessel is required to be statically pressurized with water to an overload condition of not less than 1.25 times design pressure. The ACNGS vessel was tested at 1600 psig and no defects were found. This was done in the fabrication shop and will be done again as a pre-operational test at the site. This demonstrates that the stud bolts as well as the vessel can withstand a severe static overload condition.

Q. Besides ensuring that the stud bolts meet or exceed minimum design standards for mechanical, structural and chemical properties, what else must be done to assure that the bolts are properly designed?

A. A stress analysis of the bolts must be done to ensure that the bolts will be able to carry the imposed loadings. Subarticle NB-3230 of Section III of the ASME Code

governs the stress analysis for the bolts.

Q. Referring to subpart (a) of the contention, has ATWS been considered in the design of the RPV stud bolts?

A. General Electric has included an event equivalent in all pertinent respects to the ATWS event in the reactor design. The equivalent event is the "Reactor Overpressure with Delayed Scram" and is documented in the Reactor Cycle Drawing, GE Document #762E458. For ATWS and delayed scram the peak pressure in the upper region is 1460 psig.

Q. For ATWS, what is the maximum bolt stress at the periphery and the average service stress and how does this compare to the Code allowables for an ATWS event?

A. For ATWS, the maximum bolt stress is 64 KSI, while the average service stress is 41.4 KSI. These stresses occur at a temperature of 575°F. The respective ASME Code limits are 98 KSI and 72.6 KSI at 575°F. The stresses due to ATWS are well within the ASME Code allowables and are considerably below minimum yield strength.

Q. Referring to subpart (b) of the contention, does ATWS impose the highest primary tensile stress on the stud bolts.

A. Yes.

Q. Referring to subpart (c) of the contention, will the failure of one stud bolt cause the failure of all the stud bolts?

A. No. ACNGS has 72 stud bolts each with a cross-sectional area 27.97 in^2. The total bolt crosssectional area is 2014 in^2. This is the area which was used in the stress analysis. For the actual loads which are imposed on the bolts, the minimum crosssectional area needed to carry the load is 1611.5 in^2. This equals approximately 58 bolts. In short, it is extremely improbable that the failure of one bolt will cause a failure of all the other bolts, as alleged in the contention.

Q. Can you summarize your testimony?

A. In summary, the bolt design meets all the requirements of the ASME Code and Reg. Guide 1.65. The bolting material is a high strength alloy material with excellent fracture toughness properties and is inherently resistant to stress corrosion cracking. No metallic platings are used. The mechanical properties are extensively tested to ensure the minimum acceptable properties of the material. The bolts are also extensively inspected during fabrication and in service. All stresses in the bolts are within the conservative ASME Code allowables.

RESUME

NAME: Rey L. Call

ADDRESS: 4640 Royal Grove Court, San Jose, CA 95135

BIRTHDATE: 11 December 1921

BIRTHPLACE: Bountiful, Utah, USA

PROFFSSIONAL REGISTRATION: Mechanical, California, since 1965

SOCIETY MEMBERSHIP: Member, American Society of Mechanical Engineers (ASME)

EDUCATION: BSME, University of Utah, 1950
MSME, University of Idaho, 1961

EMPLOYER: General Electric Co.
175 Curtner Avenue
San Jose, CA 95125

TITLE: Principal Engineer
Availability Engineering

WORK EXPERIENCE:

4/80 to Present — General Electric Co., San Jose, CA

Principal Engineer - Availability Engineering
Program Manager for Failure Modes and Effects Analysis preparation for a nuclear reactor project. Prepare reliability and availability studies of reactor components and systems.

1/80 to 4/80 — General Electric Co., San Jose, CA

Principal Engineer - RPV & Internals Design
Certified RPV design specifications in accordance with ASME B&PV Code, Section III Reviewed and approved vendor design drawings and stress analyses. Member, Work Group - Vessels, Section III of ASME B&PV Code Committee.

7/75 to 1/80 — General Electric Co., San Jose CA

Manager - RPV Design Unit
Responsible for mechanical design of reactor pressure vessels. Planned and performed reactor pressure vessel design to the level of detail necessary to control the design. Utilized design components and tools provided by others. Prepared design drawings and specifications required for manufacture and procurement of reactor pressure vessels. Maintained the standard designs. Provided interfaces for construction, erection, testing and operation of the reactor pressure vessels.

7/72 to 7/75 — CBI Nuclear Co., Memphis, TN

Supervisor-Development Group

Responsible for development work within Engineering in CBI Nuclear. Supervised two engineers in their work on development contracts. Reviewed proposals for development work, estimated costs and prepared Application & Authorization forms for approval of development contracts.

10/66 to 7/72 — General Electric Co., San Jose, CA

Senior Engineer

Prepared reactor pressure vessel design specifications. Certified these specifications in accordance with Section III, ASME B&PV Code. Developed reactor vessel thermal cycle drawings for the design specs. Reviewed vendor drawings and stress analyses. Designed reactor core structures.

8/62 to 10/66 — General Electric Co., San Jose, CA

Engineer I - Irradiation Processing Operation, Vallecitos, CA)

Designed irradiation capsules for testing in GETR and MTR in Idaho. Designed structures and mechanical equipment for sampling stations for an experimental power reactor. Was lead engineer in design hardware (including pressure vessel) for a plutonium subcritical experiment. Supervised work of three engineers.

9/52 to 8/62 — General Electric Co., Richland, WA

Engineer II & Engineer I

Designed and developed mechanical equipment for nuclear reactors. Monitored the reactor operations at two reactor sites to audit compliance with engineering standards and specifications. Wrote engineering standards and specifications for reactor operation. Performed design and development work on reactor fuel elements. Designed and conducted irradiation testing of reactor fuel elements.

9/45 to 9/46 — U. S. Air Force (Ogden Air Materiel Command, Ogden, UT)

Aircraft Mechanic

MILITARY SERVICE:

9/50 to 9/52 — U. S. Army

Assistant Battalion Survey Officer in a field artillery observation battalion.

7/42 to 9/45 — U. S. Navy

AMM2/C - Aircraft Mechanic