ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
UNITED STATES ATOMIC ENERGY COMMISSION
WASHINGTON, D.C. 20545

November 19, 1968

Dr. W. R. Stratton
Dr. S. H. Bush
Mr. H. Etherington
Mr. N. J. Palladino
Dr. S. H. Hanauer
Dr. J. M. Hendrie

JERSEY CENTRAL POWER & LIGHT COMPANY - OYSTER CREEK UNIT NO. 1

Information provided to the applicant concerning the Oyster Creek Unit No. 1 Subcommittee meetings on November 22 and 23, 1968 is as follows:

<u>Friday - November 22, 1968</u>

A. The applicant should be prepared to give a presentation concerning:

1. The Reactor Pressure Vessel
2. The Operating Staff

B. The applicant should be prepared to discuss the following:

1. Pressure Vessel Repairs
2. Primary System Leak Detection and Procedures When Leaks are Detected
3. Technical Specifications
4. Comparison of In-Service Inspection with Proposed N-45 Code
5. Operating Staff, Including Training, Experience, Organization, Support by GE, etc.
6. Reactor Building Closed Cooling Water System
7. Radiolysis and the Advantages of Inerting
8. Variability of Flux Trip Point with Flow
9. Feedwater Control Valves on FWCIS

<u>Saturday - November 23, 1968</u>

The applicant should be prepared to discuss:

1. Subchannel Separation (Including Physical Separation)
2. Instrumentation-General Considerations
3. Auto-relief, A-C Interlock and Information Available to Operator
4. Cable Tray Overloading

Marvin C. Gaske

Marvin C. Gaske, Assistant
to Executive Secretary

3/19/284

October 10, 1968

W. R. Stratton, Oyster Creek Subcommittee Chairman

OYSTER CREEK - REACTOR PRESSURE VESSELS

[illegible] n Stub-Tube Fabrication Stress as Discussed at DRL Meeting of Aug [illegible] 968.

[illegible] level strain at the field weld is inevitable and it would [illegible] be useful to attempt to control induced stresses remote [illegible].

This position is refuted by the observation that stub-tube cracking did not occur adjacent to the field weld but did occur remote from the field weld. The stress at the outer surface of the stub tubes is believed to be compressive near the weld and tensile where cracking occurred.

2. The magnitude of the stress, above some low threshold value, is not an important factor in stress corrosion.

(a) There appeared to be conflicting views on this.

(b) Again the position is not supported by the behavior of the stub tubes, which were highly stressed over much of the surface but which cracked in the region of highest calculated stress.

(c) Many reports and investigations of stress corrosion (including some by GE) attribute failures in part to high stress.

3. Even if an identical stress pattern had been developed by a mechanically applied shrink ring at the top of the stub tube instead of by contraction of the weld region, no attention would have been paid to the stress.

Does this imply that the Code requirements are not considered applicable to the stub tubes after field welding? Why would the stresses not be considered secondary bending stresses subject to the 3 S_m limit?

File: Oyster Creek Unit 1

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4. Load cycling will redistribute the stress and a high initial stress is not important when an alternating stress is superimposed.

There is no question that redistribution occurs, but Section N-415.2, Design for Cyclic Loading, explicitly states that compliance with fatigue requirements does not exempt from meeting the general stress limits of Section N-414.

5. (a) Buttering with 308 L eliminates the hazard of stress corrosion in Oyster Creek. (b) The stub tubes in Nine Mile Point have not been pre-exposed to a damaging environment and will not be susceptible to stress corrosion in service - they will not be coated with 308 L.

Both presumptions are probably valid, but the frequency of unexpected stress-corrosion failures leaves room for concern. Stress corrosion of austenitic steel has been with us for over 40 years, and over most of that time the phenomenon was currently believed to be reasonably well understood. Yet unexpected failures are continuing to occur. The Dresden 1 piping failures provide examples:

(a) First, stress corrosion was found in the HAZ of pipe welds, and the Battelle report describes these failures as "unexpected" and with "no precedent" in the high purity water environment.

(b) Then, stress corrosion was found in a straight run of unsensitized (Q.A.) pipe. Battelle again reported no precedent and no good explanation.

(c) In a somewhat different category, stress corrosion failures were observed in cap screws of fuel channels.

In such cases, it is usual to point to some anomalous condition that might have contributed to failure, but never with any suggestion that the failure should have been predicted.

The recent work by GE and others on the effects of pre-service environment is impressive, but history suggests that we are not yet in a position to be either complacent or dogmatic on the subject of stress corrosion.

6. A mandrel has been used to reweld the control rod housings to the stub tubes. The only purpose was to maintain necessary clearance for thermal sleeves. Mitigation of the stressed condition in the stub tubes was not an objective and was not considered. Use of a mandrel may increase the contraction strain in the weld metal, but since there is already a 5% strain, any additional strain is unimportant.

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The 2% is the algebraic sum of free thermal contraction and stress induced strain. We do not know what the true strain is in the weld, but it is probably much less than 2%. Complete constraint would increase the unknown tensile strain by 2%.

The 10% ferrite in a 308 L weld should provide ample protection against micro-cracking and hot cracks in an unconstrained weld. What evidence is there that this is also true for a weld rigidly constrained against shrinkage?

Harold Etherington
ACRS Member

CC: Dr. D. Okrent

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MCDB:bwd
10/16/68

Project: Oyster Creek Unit No. 1

Status : Provisional Operating License Requested

Background: The Oyster Creek Subcommittee met with representatives of the applicant on June 22, October 20, and November 17 and 27, 1967 in Washington, D.C. and on July 28, 1967 in Toms River, New Jersey. The project was considered at the November and December 1967 ACRS meetings.

DRL Report

DRL has concluded that there are two areas that must be resolved with the applicant prior to licensing. These are:

1. Provision of an interlock in the auto-relief system to prevent blowdown unless availability of the core spray system is assured.

2. The technical bases upon which plant turnover from GE to Jersey Central must be resolved. (DRL also needs to determine that the reactor vessel repairs are adequate, and an acceptable set of Technical Specifications needs to be developed.)

DRL has identified the following which will be studied after initial licensing:

1. Improved primary system leak detection and the in-service inspection and surveillance programs.

2. Detailed review of design of FWCI prior to system operation.

3. Continued review of the design and performance capability of the main steam line isolation values.

DRL currently plans to provide a report to the Committee in early November concerning the reactor pressure vessel and quality assurance and control.

Guidance Provided to Applicant

The applicant was informed regarding the October 17, 1968 Subcommittee meeting:

A. The reactor pressure vessel would not be discussed.

B. A presentation should be given regarding a general review of the ECCS and the emergency power supplies.

C. The applicant should be prepared to discuss:

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1. The emergency condenser isolation valves.

2. The auto-relief system.

3. Primary system leak detection during reactor operation.

4. Main steam line valve tests.

5. Instrumentation, including sub-channel separation and flow signal to APRM scrams (including need for redundancy).

6. Emergency plans.

7. Startup and power ascension program.

8. Technical specifications.

   (a) Safety limits

   (b) In-service inspection and surveillance

Questions

1. DRL has indicated several items they plan to report at the November meeting of the ACRS. It might be worthwhile for the Subcommittee to hear the status of these items:

   A. Coastal Engineering Research Center's evaluation of the maximum probable flood height.

   B. The need for vibrational testing of the steam separator.

   C. Status of the Technical Sepcifications.

2. The acceptability of using a 120% of full power value for the overpower set point, instead of a smaller value, might be discussed.

3. The Nine Mile Point reactor has the same problem regarding the location of the isolation valves in the line to the emergency condenser at a point outside the drywell. DRL might be asked if they would also propose to accept this situation for the Nine Mile Point Reactor.

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