

TENNESSEE VALLEY AUTHORITY

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NOV 25 1987

WBRD-50-390/87-20
WBRD-50-391/87-24

10 CFR 50.55(e)

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, D.C. 20555

Gentlemen:

In the Matter of the Application of)
Tennessee Valley Authority)

Docket Nos. 50-390
50-391

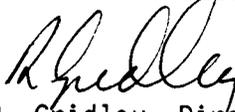
WATTS BAR NUCLEAR PLANT (WBN) UNITS 1 AND 2 - DEFICIENCIES IN THE DESIGN OF
THE HYDROGEN RECOMBINER ACCESS PLATFORM AND STEAM GENERATOR LOWER LATERAL
SUPPORT - WBRD-50-390/87-20 AND WBRD-50-391/87-24 - INTERIM REPORT

The subject deficiency was initially reported to NRC-Region II Inspector
Steve Elrod on October 22, 1987, in accordance with 10 CFR 50.55(e) as
CAQRs WBP 870759 and WBP 870760 for unit 1. CAQRs WBP 871094 and WBP 871174
were subsequently initiated to document the deficiency for unit 2. Enclosed
is our interim report. We expect to submit our final report on or about
April 30, 1989.

If there are any questions, please telephone R. D. Schulz at (615) 365-8527.

Very truly yours,

TENNESSEE VALLEY AUTHORITY


R. Gridley, Director
Nuclear Licensing and
Regulatory Affairs

Enclosure
cc: See page 2

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U.S. Nuclear Regulatory Commission

NOV 25 1987

cc (Enclosure):

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ENCLOSURE

WATTS BAR NUCLEAR PLANT UNIT 1 AND 2
DEFICIENCIES IN THE DESIGN OF THE HYDROGEN RECOMBINER
ACCESS PLATFORM AND STEAM GENERATOR LOWER LATERAL SUPPORT
WBRD-50-390/87-20 AND WBRD-50-391/87-24
CAQRs WBP 870759, WBP 870760, WBP 871084 AND WBP 871174
10 CFR 50.55(e)

INTERIM REPORT

Description of Deficiency

Deficiencies have been identified with the design calculations performed for the hydrogen recombiner access platforms and the lower steam generator lateral supports. These structures were designed with the help of the Georgia Tech Structural Analysis Design Language computer program (GT STRUDL). This is a revision to the STRUDL program originally written at the Massachusetts Institute of Technology. Although the GT STRUDL computer program itself was adequate, the output from this computer program may not have been correctly interpreted and incorporated into the design of the structures. As a result, these structures may be stressed beyond the allowable limits.

The items listed below represent the significant deficiencies identified during a review of the structures for additional attachments.

For the hydrogen recombiner access platform:

- Default values for GT STRUDL 69AISC code check parameters (such as the effective length factor and member minimum yield strength) were used for all members. The automatically generated values may not correctly specify the values for an accurate analysis and may not be conservative.
- Deflections were not checked in the design. A later analysis indicates a deflection of approximately 4.6 inches at one corner of the structure.

Because of the above deficiencies, the computed stresses may be unconservative and, consequently, some structural steel details for weld sizes and connections may not have adequate evaluations.

For the steam generator lower lateral supports:

- GT STRUDL 69AISC code checks were used for stress analysis without including torsional stresses in the stress combination.

Safety Implications

These structures, which were designed with improper evaluations of the structural members and their associated connections, could be overstressed. This could lead to failure of the structure/support during a seismic event.

The hydrogen recombiner access platform is located inside containment about 22 feet above the refueling deck and holds one of the two hydrogen recombiners. The main concern with this platform is that it may fall during a seismic event and damage other safety-related equipment. Two containment isolation valves and lines (one supply line and one return line for the ice condenser floor cooling system) are located under this platform and could fail if the platform strikes them.

The lower steam generator support provides only lateral support of the steam generator in the event of a main steam line break. However, it does provide primary support to safety-related piping which is attached to it. Should this support fail, then various safety-related pipes could fail.

Either of the above two failure modes could adversely affect safe operation of the plant.

Interim Progress

The deficiency appears to have resulted from inadequate design criteria. TVA presently plans to revise the design criteria to ensure that requirements are adequately specified. Based on the appropriate design criteria requirements, TVA will evaluate the subject structures to determine if they are adequate and will rework if necessary.

TVA also plans to evaluate other similar structures to determine the extent of this deficiency. A final report which addresses the extent of the deficiency and any additional corrective actions will be submitted to NRC on or about April 30, 1989.