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JUL 19 1982

Docket No: 50-369

Mr. William O. Parker, Jr.
Vice President - Steam Production
Duke Power Company
P.O. Box 33189
422 South Church Street
Charlotte, North Carolina 28242

Dear Mr. Parker:

Subject: Continued Operation of Unit 1 (McGuire Nuclear Station)

By letter dated May 20, 1982, we informed you that your McGuire 1 operating program was acceptable for a period extending from May to July 4, 1982, including operation at 75% power for a maximum of 720 hours (30 days). On June 23, 1982, Unit 1 was shutdown after 720 hours of operation at 75% power.

Subsequent to the June 23, 1982 shutdown of Unit 1 for steam generator ECT inspection, you informed us that inspection of the Unit 1 reactor coolant system thermal sleeve connection areas revealed that loop B thermal sleeve was detached and missing. Your letter report on thermal sleeves dated July 13, 1982, presents information and evaluation of this occurrence and situation and proposes to restart Unit 1 with the detached thermal sleeve in the RCS probably in the lower reactor internals area. Included in your proposal you have voluntarily elected to increase the frequency of certain technical specification surveillance requirements until such time that the affected thermal sleeves are removed from the RCS. These surveillance matters are further detailed in your July 14, 1982 letter to the staff.

In addition to the documents filed by you, we have had the benefit of meeting with you and Westinghouse on July 14, 1982, to discuss the result of your findings.

Based on our review of the information provided and the discussions held on July 14, 1982, we conclude that continued operation of McGuire Unit 1 until the next refueling outage or outage of sufficient duration, with the thermal sleeves as presently positioned is acceptable and will not result in undue risk to the health and safety of the public. A summary of the results of our review is presented in the enclosure No. 1.

OFFICE							
SURNAME							
DATE							

Your letter on results of steam generator tube testing dated July 13, 1982 provided the staff with preliminary results of the eddy current testing (ECT) of all steam generators. You stated that a detailed report describing the results of the June 1982 ECT would be submitted by July 30, 1982, and that operation of Unit 1 at 50% power in the interim is prudent pending final evaluation of the ECT. As a result of our review of the information provided and evaluation by our consultant, we conclude and find acceptable your McGuire 1 operating program for interim operation at 50% power until completion of the staff's evaluation of your final ECT report.

We require that the staff be immediately notified in the event there is any indication of steam generator behavior contrary to the information which was provided in your July 13, 1982 letter. A summary of the results of our review is presented in the enclosure No. 2.

Sincerely,

/S/

Darrell G. Eisenhut, Director
Division of Licensing

Enclosure:
As stated

cc: See next page

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*GLainas	RTedesco
*7/18/82	7/16 /82

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Docket No: 50-369

Mr. William O. Parker, Jr.
Vice President - Steam Production
Duke Power Company
P.O. Box 33189
422 South Church Street
Charlotte, North Carolina 28242

Dear Mr. Parker:

Subject: Continued Operation of Unit 1 (McGuire Nuclear Station)

By letter dated May 20, 1982, we informed you that your McGuire 1 operating program was acceptable for a period extending from May to July 4, 1982, including operation at 75% power for a maximum of 720 hours (30 days). On June 23, 1982, Unit 1 was shutdown after 720 hours of operation at 75% power.

Your letter dated July 13, 1982 provided the staff with preliminary results of the eddy currents testing (ECT) of all steam generators. You stated that a detailed report describing the results of the June 1982 ECT would be submitted by July 30, 1982, and that operation of Unit 1 at 50% power in the interim is prudent pending final evaluation of the ECT. As a result of our review of the information provided and evaluation by our consultant, we conclude and find acceptable your McGuire 1 operating program for interim operation at 50% power until completion of the staff's evaluation of your final ECT report.

We require that the staff be immediately notified in the event there is any indication of steam generator behavior contrary to the information which was provided in your July 13, 1982 letter. A summary of the results of review is presented in the enclosure.

Sincerely,

Robert Purple, Deputy Director
Division of Licensing

Enclosure:
As stated

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Docekt No: 50-369

Mr. William O. Parker, Jr.
Vice President - Steam Production
Duke Power Company
P.O. Box 33189
422 South Church Street
Charlotte, North Carolina 28242

Dear Mr. Parker:

Subject: Operation With Detached Thermal Sleeve (McGuire Nuclear Station, Unit 1)

Subsequent to the June 23, 1982 shutdown of Unit 1 for steam generator ECT inspection, you informed us that inspection of the Unit 1 reactor coolant system thermal sleeve connection areas revealed that loop B thermal sleeve was detached and missing. Your letter report dated July 13, 1982, presents information and evaluation of this occurance and situation and proposes to restart Unit 1 with the detached thermal sleeve in the RCS probably in the lower reactor internals area. Included in your proposal you have voluntarily elected to increase the frequency of certain technical specification surveillance requirements until such time that the affected thermal sleeves are removed from the RCS. These surveillance matters are further detailed in your July 14, 1982 letter to the staff.

In addition to the documents filed by you, we have had the benefit of meeting with you and Westinghouse on July 14, 1982, to discuss the result of your findings.

Based on our review of the information provided and the discussions held on July 14, 1982, we conclude that continued operation of McGuire Unit 1 until the next refueling outage or outage of sufficient duration, with the thermal sleeves as presently positioned is acceptable and will not result in undue risk to the health and safety of the public. A summary of the results of our review is presented in the enclosure.

Sincerely,

Original signed by
Darrell G. Eisenhut

Darrell G. Eisenhut, Director
Division of Licensing

Enclosure:
As stated

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McGuire

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James P. O'Reilly, Regional Administrator
U.S. Nuclear Regulatory Commission,
Region II
101 Marietta Street, Suite 3100
Atlanta, Georgia 30303

ENCLOSURE 1

REVIEW OF PROPOSED OPERATION WITH DETACHED THERMAL SLEEVE McGUIRE NUCLEAR STATION, UNIT 1

Ref: (1) Duke Power Company letter report dated July 13, 1982 and letter dated July 14, 1982
(2) Licensee meeting of July 14, 1982

INTRODUCTION

The McGuire Nuclear Station, Unit 1, was shutdown on June 23, 1982, for purposes of eddy current testing of all Model D steam generators. Pursuant to recent evidence of the degradation of thermal sleeve components in the reactor coolant system of the Trojan plant, another Westinghouse plant, the licensee promptly initiated an inspection of all thermal sleeves in the Unit 1 reactor coolant system utilizing radiography techniques. The RCS contains the following thermal sleeves: (4) -10" accumulator nozzle cold leg, (2) -3" charging nozzle cold legs and (1) -14" pressurizer surge nozzle hot leg.

BACKGROUND

The radiograph of the 10" accumulator nozzle thermal sleeve on loop B revealed that it was detached and missing. This was confirmed by the licensee by a visual inspection with a small TV camera going through the upstream check valve on the 10" line. The licensee has inspected all other connections to the RCS having similar thermal sleeves to assure that the remaining sleeves were in place with their welds intact. Only one detached (B loop) thermal sleeve was evidenced by the licensee's inspection. The licensee reported that the loop B 10" line welds located at the top of the sleeve had failed at the interface to the nozzle wall with possibly only a small portion of one weld remaining on the nozzle wall. The nozzle wall, according to the licensee, showed no indication that the sleeve had broken apart prior to being released. As a result of monitoring of the lower reactor vessel area for loose parts during low flow and normal RCS flow conditions, the licensee concludes that the missing 10" thermal sleeve is located in the lower reactor internals and that the small impacts during low flow conditions are of a minor nature. No movement is indicated when full flow is present. The licensee considers this to indicate that the sleeve is parked and remains fixed in place.

EVALUATION AND CONCLUSIONS

The staff has reviewed the information provided by the licensee in its letters dated July 13 and July 14, 1982, and during the July 14, 1982 meeting with the licensee and Westinghouse.

A detailed stress analysis was performed by Westinghouse for the accumulator and other nozzles without thermal sleeves. Finite element techniques were used to calculate thermal stress effects under the most adverse operating transients, including their contribution to cumulative fatigue damage. The analysis included the whole nozzle structure and the welds to the connecting pipe. In addition to the operating transients, all other mechanical loads were included. The analytical results indicated that the stress and fatigue damage are within the allowable limits set by subsection NB of section III of the ASME Code. The analysis and results are acceptable to verify nozzle integrity. It is the licensee's intent to remove all remaining in-place nozzle thermal sleeves and recover any sleeves which are no longer in place at the first refueling outage or the first extended outage period. It was noted that Westinghouse plants subsequent to Comanche Peak station omit these sleeves in the RCS.

The impact and wedging effects of a loose thermal sleeve on reactor internals, steam generators, and primary system pipe have been evaluated by the licensee. We agree with the results of the evaluation that such effects are unlikely either to impair the reactor coolant pressure boundary or to cause unacceptable safety concerns due to the limited available impact energy which can be imparted on randomly targeted mechanical components.

The potential hydraulic effects of the detached thermal sleeve, including blockage effects for normal operation, transients and accidents, was evaluated by the licensee who indicated that the amount of total RCS flow degradation due to blockage would be less than 0.6 percent, with a resultant flow still greater than design flow. The licensee stated that initial conditions for postulated transients would be substantially unaltered by this amount of blockage.

In his analysis the licensee has indicated that the loose sleeve could damage an instrument thimble in the lower plenum of the reactor vessel, resulting in RCS leakage. The complete rupture of three such thimbles would be necessary to result in loss of RCS inventory at a rate greater than the makeup rate of a charging pump. Leakage from one damaged thimble would be detected by normal identified leakage detection equipment, and leakage response procedures would be followed. This scenario falls within the design envelope which was reviewed and found acceptable during the operating license review stage. The licensee also discussed the effects of their debris (thermal sleeve) on the postulated design basis LOCA analysis for McGuire stating that calculated results would not change relative to the FSAR analyses. In addition the licensee agreed to the following operational procedure requirements:

1. Control rod operability checks will be performed weekly. The Technical Specification requirement is monthly.

2. Incore flux mapping will be performed weekly. The Technical Specification requirement is monthly.

3. Analysis to determine reactor coolant activitiy will be performed daily. The Technical Specification requirement is weekly. These results will be reviewed for trends.

4. The Loose Parts Monitoring System will be checked by verification of audio signal by the Shift Technical Advisor on an hourly frequency.

5. The plant will commence a controlled shutdown if the Loose Parts Monitoring System is inoperable for more than 72 hours.

6. At the time of an extended outage to remove the existing thermal sleeves, an ultrasonic examination of the affected areas of the nozzles will be performed.

We therefore conclude that continued operation of McGuire unit 1 until the next refueling outage or outage of sufficient duration with the thermal sleeves as presently positioned is acceptable without undue risk to the health and safety of the public.

REVIEW OF McGUIRE UNIT 1 PRELIMINARY EDDY CURRENT RESULTS OF MODEL D STEAM GENERATOR AFTER OPERATION AT 75% POWER

REF: Duke Power Company Letter Report dated July 13, 1982

INTRODUCTION

By letter dated July 13, 1982, Duke Power Company submitted preliminary results of the eddy current inspection (ECT) of their steam generators, and based on these results proposed to return to power operation at 50% power level. The licensee intends to submit a detailed report describing the results of the ECT by July 30, 1982 and that operation at 50% power level in this interim period is prudent pending final evaluation of the ECT.

The licensee concluded that the plant can be operated at 50% power with no deleterious effect on the steam generator tubes due to fretting wear. This conclusion was based on the following:

1. Results of the ECT conducted after 50% power operation which revealed no detectable tube degradation (Nov. 1981) and;

2. Results of ECT conducted at Almaraz after 1500 hours at 50% power operation which revealed no significant wear.

DISCUSSION

McGuire 1 had accumulated the following operating history at the time it was shutdown on February 26, 1982:

Power Level	Hours at or above this power level
50%	1500
75%	324
90%	72
100%	23

The total number of effective full power hours at that time was 1093.

On March 14, 1982 McGuire 1 commenced operation initially at 50% power for 1500 hours and then at 75% power for 720 hours until its shutdown for inspection on June 23, 1982. Eddy current testing during the March 1982 outage revealed four tubes with O.D. indications in steam generator "C". These indications have been attributed to small volume wear defects which the licensee estimated to have a conservative upper bound of 4.0×10^{-4} cubic inches for the volume of the largest defect. Eddy current analyses indicated that the wall penetrations were $<$ 20% for these defects.

In his submittal on April 28, 1982 to justify operation for 720 hours at 75% power the licensee estimated that tube wear due to fretting during this period would be equivalent to a volume of 8.89×10^{-4} cubic inches. Based on this incremental tube wear, the four worst tubes in steam generator "C" would have an estimated defect volume of 1.29×10^{-3} cubic inches; which is equivalent to a defect of less than 10 mils (23% penetration).

The preliminary ECT results presented in the July 13, 1982 submittal reveals the following indications:

"A" Steam Generator

- Indications were observed on 15 tubes. Maximum depth was approximately 15 percent. Affected tubes are the same tubes which showed distorted support plate signals during the previous ECT in March, 1982.

"B" Steam Generator

No indications.

"C" Steam Generator

Eight indications observed on 6 tubes. The largest indication was on tube R49C 40. This tube previously had an indication called <20 percent. The indication has grown to approximately 23 percent which is consistent with wear rate estimates previously made.

"D" Steam Generator

No indications.

The above ECT evaluations were based on absolute single coil probe measurements, deemed to be the most effective for measuring wear volumes. One tube in the "A" steam generator was plugged based on preliminary evaluation of the differential ECT data. This evaluation indicated 46 percent through wall wear. It was suspected that this was an over-estimation of the wear, but rather than wait until final evaluation of the absolute ECT data, the decision was made to plug the tube to avoid schedule delay.

The licensee is presently evaluating the ECT data in detail in order to determine a plan for the next period of operation.

EVALUATION AND CONCLUSION

We find that McGuire may be operated at power levels not to exceed 50% from the date of restart following the most recent steam generator inspection until the staff's evaluation of the licensee's final report to be submitted by July 30, 1982 is completed, without undue risk to public health and safety. This finding is based upon the following:

(1) Up through February 26, 1982, McGuire had operated for 1500 hours at power levels at and above 50%, including 324 hours at and above 75% power while incurring only minor wear (conservatively estimated at 4.0×10^{-4} cubic inches and $< 20\%$ penetration) on four tubes.;

(2) During the period of March 14, 1982 through June 23, 1982 after 1500 hours at 50% power and 720 hours at 75% power-additional wear on the previously degraded tubes was consistent with wear rate estimates based on operation at 75% power. Hence, no wear or further degradation is attributable to operation at 50% power. Our consultant, Dr. C. V. Dodd of ORNL, verified that the preliminary ECT data indicates that there was negligible defect growth during this last period of operation;

(3) Results from Almaraz after 1500 hours at 50% power indicate no significant tube wear based on ECT data; and

(4) Restrictive limits on allowable primary to secondary leakage in the the Technical Specifications provide adequate assurance of tube integrity.

We request that the NRC staff be immediately notified in the event that further evaluation of the ECT data indicates that steam generator tube behaviour is contrary to the information which was provided in the July 13, 1982 submittal.