

September 21, 1987

Docket No. 50-364

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Mr. R. P. McDonald  
Senior Vice President  
Alabama Power Company  
Post Office Box 2641  
Birmingham, Alabama 35291-0400

Dear Mr. McDonald:

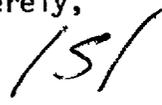
SUBJECT: ISSUANCE OF AMENDMENT NO. 64 TO FACILITY OPERATING LICENSE  
NO. NPF-8 -JOSEPH M. FARLEY NUCLEAR PLANT, UNIT 2, REGARDING  
TESTS OF STEAM GENERATOR TUBES IN THE TUBESHEET REGION  
(TAC NO. 65287)

The Nuclear Regulatory Commission has issued the enclosed Amendment No. to Facility Operating License NPF-8 for the Joseph M. Farley Nuclear Plant, Unit 2. The amendment consists of changes to the Technical Specifications in response to your application transmitted by letter dated May 4, 1987.

The amendment modifies Technical Specification (TS) 3/4.4.6 surveillance and reporting requirements for testing of the steam generator tubes in the tube-sheet region. The TS changes identified as F\*(F-star) delete the definition of plugging or repairing for defects below F\*, but require reporting of F\* conditions.

A copy of our related Safety Evaluation is also enclosed. The Notice of Issuance will be included in the Commission's regular bi-weekly Federal Register notice.

Sincerely,



Edward A. Reeves, Project Manager  
Project Directorate II-1  
Division of Reactor Projects I/II

8709250240 870921  
PDR ADDCK 05000364  
P PDR

Enclosures:

1. Amendment No. 64 to NPF-8
3. Safety Evaluation

cc: w/enclosures:  
See next page

LA:PD21:DRPR  
PAnderson  
8/11/87

PM:PD21:DRPR  
EReesves/dsf  
8/17/87

D:PD21:DRPR  
EAdensam  
9/21/87

Mr. R. P. McDonald  
Alabama Power Company

Joseph M. Farley Nuclear Plant

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Resident Inspector  
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AMENDMENT NO. 64 TO FACILITY OPERATING LICENSE NO. NPR-8 - FARLEY, UNIT 2

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JOSEPH M. FARLEY NUCLEAR PLANT, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 64  
License No. NPF-8

1. The Nuclear Regulatory Commission (the Commission) has found that:
  - A. The application for amendment by Alabama Power Company (the licensee), dated May 4, 1987, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;
  - B. The facility will operate in conformity with the application, the provisions of the Act, and the regulations of the Commission;
  - C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;
  - D. The issuance of this license amendment will not be inimical to the common defense and security or to the health and safety of the public; and
  - E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.
2. Accordingly, the license is amended by changes to the Technical Specifications, as indicated in the attachment to this license amendment; and paragraph 2.C.(2) of Facility Operating License No. NPF-8 is hereby amended to read as follows:

8709250244 870921  
PDR ADOCK 05000364  
P PDR

(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 64, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of its date of issuance and shall be implemented within 30 days of receipt of this amendment.

FOR THE NUCLEAR REGULATORY COMMISSION

*151*

Elinor G. Adensam, Director  
Project Directorate II-1  
Division of Reactor Projects I/II

Attachment:  
Changes to the Technical  
Specifications

Date of Issuance: September 21, 1987

*[Signature]*  
LA: PD21:DRPR  
PAnderson  
8/17/87

*[Signature]*  
PM: PD21:DRPR  
EReeves/dsf  
8/17/87

*[Signature]*  
OGC-B  
8/19/87

*[Signature]*  
D: PD21:DRPR  
EAdensam  
9/21/87

ATTACHMENT TO LICENSE AMENDMENT

AMENDMENT NO. 64 FACILITY OPERATING LICENSE NO. NPF-8

DOCKET NO. 50-364

Replace the following pages of the Appendix A Technical Specifications with the enclosed pages as indicated. The revised pages are identified by amendment number and contain vertical lines indicating the area of change.

Remove Pages

3/4 4-9  
3/4 4-10  
3/4 4-12  
3/4 4-13  
--  
3/4 4-15  
B3/4 4-3a

Insert Pages

3/4 4-9  
3/4 4-10  
3/4 4-12  
3/4 4-13  
3/4 4-13a  
3/4 4-15  
B3/4 4-3a

## REACTOR COOLANT SYSTEM

### 3/4.4.6 STEAM GENERATORS

#### LIMITING CONDITION FOR OPERATION

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3.4.6 Each steam generator shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With one or more steam generators inoperable, restore the inoperable generator(s) to OPERABLE status prior to increasing  $T_{avg}$  above 200°F.

#### SURVEILLANCE REQUIREMENTS

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4.4.6.0 Each steam generator shall be demonstrated OPERABLE by performance of the following augmented inservice inspection program and the requirements of Specification 4.0.5.

4.4.6.1 Steam Generator Sample Selection and Inspection - Each steam generator shall be determined OPERABLE during shutdown by selecting and inspecting at least the minimum number of steam generators specified in Table 4.4-1.

4.4.6.2.1 Steam Generator Tube<sup>#</sup> Sample Selection and Inspection - The steam generator tube minimum sample size, inspection result classification, and the corresponding action required shall be as specified in Table 4.4-2. The inservice inspection of steam generator tubes shall be performed at the frequencies specified in Specification 4.4.6.3 and the inspected tubes shall be verified acceptable per the acceptance criteria of Specification 4.4.6.4. The tubes selected for each inservice inspection shall include at least 3% of the total number of tubes in all steam generators. Selection of tubes to be inspected is not affected by the F\* designation. When applying the exceptions of 4.4.6.2.1.a through 4.4.6.2.1.c, previous defects or imperfections in the area repaired by sleeving are not considered an area requiring reinspection. The tubes selected for these inspections shall be selected on a random basis except:

- a. Where experience in similar plants with similar water chemistry indicates critical areas to be inspected, then at least 50% of the tubes inspected shall be from these critical areas.
- b. The first sample of tubes selected for each inservice inspection (subsequent to the preservice inspection) of each steam generator shall include:
  1. All nonplugged tubes that previously had detectable wall penetrations greater than 20%.

<sup>#</sup>When referring to a steam generator tube, the sleeve shall be considered a part of the tube if the tube has been repaired per Specification 4.4.6.4.a.9.

## REACTOR COOLANT SYSTEM

### SURVEILLANCE REQUIREMENTS (Continued)

2. Tubes in those areas where experience has indicated potential problems.
  3. At least 3% of the total number of sleeved tubes in all three steam generators or all of the sleeved tubes in the generator chosen for the inspection program, whichever is less. These inspections will include both the tube and the sleeve.
  4. A tube inspection (pursuant to Specification 4.4.6.4.a.8) shall be performed on each selected tube. If any selected tube does not permit the passage of the eddy current probe for a tube or sleeve inspection, this shall be recorded and an adjacent tube shall be selected and subjected to a tube inspection.
- c. The tubes selected as the second and third samples (if required by Table 4.4-2) during each inservice inspection may be subjected to a partial tube inspection provided:
1. The tubes selected for these samples include the tubes from those areas of the tube sheet array where tubes with imperfections were previously found.
  2. The inspections include those portions of the tubes where imperfections were previously found.

The results of each sample inspection shall be classified into one of the following three categories:

<u>Category</u>	<u>Inspection Results</u>
C-1	Less than 5% of the total tubes inspected are degraded tubes and none of the inspected tubes are defective.
C-2	One or more tubes, but not more than 1% of the total tubes inspected are defective, or between 5% and 10% of the total tubes inspected are degraded tubes.
C-3	More than 10% of the total tubes inspected are degraded tubes or more than 1% of the inspected tubes are defective.

Note: In all inspections, previously degraded tubes or sleeves must exhibit significant (greater than 10%) further wall penetrations to be included in the above percentage calculations.

4.4.6.2.2 Steam Generator F\* Tube Inspection - In addition to the minimum sample size as determined by Specification 4.4.6.2.1, all F\* tubes will be inspected within the tubesheet region. The results of this inspection will not be a cause for additional inspections per Table 4.4-2.

## REACTOR COOLANT SYSTEM

### SURVEILLANCE REQUIREMENTS (Continued)

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#### 4.4.6.4 Acceptance Criteria

a. As used in this Specification:

1. Imperfection means an exception to the dimensions, finish or contour of a tube or sleeve from that required by fabrication drawings or specifications. Eddy-current testing indications below 20% of the nominal wall thickness, if detectable, may be considered as imperfections.
2. Degradation means a service-induced cracking, wastage, wear or general corrosion occurring on either inside or outside of a tube or sleeve.
3. Degraded Tube means a tube, including the sleeve if the tube has been repaired, that contains imperfections greater than or equal to 20% of the nominal wall thickness caused by degradation.
4. % Degradation means the percentage of the tube or sleeve wall thickness affected or removed by degradation.
5. Defect means an imperfection of such severity that it exceeds the plugging or repair limit. A tube or sleeve containing a defect is defective.
6. Plugging or Repair Limit means the imperfection depth at or beyond which the tube shall be repaired (i.e., sleeved) or removed from service by plugging and is greater than or equal to 40% of the nominal tube wall thickness. This definition does not apply to the area of the tubesheet region below the F\* distance in F\* tubes. For a tube that has been sleeved, through wall penetration of greater than or equal to 31% of sleeve nominal wall thickness in the sleeve requires the tube to be removed from service by plugging.
7. Unserviceable describes the condition of a tube or sleeve if it Teaks or contains a defect large enough to affect its structural integrity in the event of an Operating Basis Earthquake, a loss-of-coolant accident, or a steam line or feedwater line break as specified in 4.4.6.3.c, above.
8. Tube Inspection means an inspection of the steam generator tube from the point of entry (hot leg side) completely around the U-bend to the top support of the cold leg. For a tube that has been repaired by sleeving, the tube inspection should include the sleeved portion of the tube.
9. Tube Repair refers to mechanical sleeving, as described by Westinghouse report WCAP-11178 Rev. 1, which is used to maintain a tube in service or return a tube to service. This includes the removal of plugs that were installed as a corrective or preventive measure.

## REACTOR COOLANT SYSTEM

### SURVEILLANCE REQUIREMENTS (Continued)

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10. Preservice Inspection means an inspection of the full length of each tube in each steam generator performed by eddy current techniques prior to service to establish a baseline condition of the tubing. This inspection shall be performed after the field hydrostatic test and prior to initial POWER OPERATION using the equipment and techniques expected to be used during subsequent inservice inspections.
  11. F\* Distance is the distance of the expanded portion of a tube which provides a sufficient length of undegraded tube expansion to resist pullout of the tube from the tubesheet. The F\* distance is equal to 1.79 inches and is measured down from the top of the tubesheet or the bottom of the roll transition, whichever is lower in elevation.
  12. F\* Tube is a tube:
    - a) with degradation equal to or greater than 40% below the F\* distance, and b) which has no indication of imperfections greater than or equal to 20% of nominal wall thickness within the F\* distance, and c) that remains in service.
  13. Tube Expansion is that portion of a tube which has been increased in diameter by a rolling process such that no crevice exists between the outside diameter of the tube and the hole in the tubesheet.
- b. The steam generator shall be determined OPERABLE after completing the corresponding actions (plug or repair of all tubes exceeding the plugging or repair limit) required by Table 4.4-2.

#### 4.4.6.5 Reports

- a. Following each inservice inspection of steam generator tubes, the number of tubes plugged, repaired or designated F\* in each steam generator shall be reported to the Commission within 15 days of the completion of the inspection, plugging or repair effort.
- b. The complete results of the steam generator tube and sleeve inservice inspection shall be submitted to the Commission in a Special Report pursuant to Specification 6.9.2 within 12 months following the completion of the inspection. This Special Report shall include:
  1. Number and extent of tubes and sleeves inspected.
  2. Location and percent of wall-thickness penetration for each indication of an imperfection.
  3. Identification of tubes plugged or repaired.

REACTOR COOLANT SYSTEM

SURVEILLANCE REQUIREMENTS (Continued)

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- c. Results of steam generator tube inspections which fall into Category C-3 shall be considered a REPORTABLE EVENT and shall be reported pursuant to 10CFR50.73 prior to resumption of plant operation. The written report shall provide a description of investigations conducted to determine the cause of the tube degradation and corrective measures taken to prevent recurrence.

TABLE 4.4-2  
STEAM GENERATOR TUBE INSPECTION

1ST SAMPLE INSPECTION			2ND SAMPLE INSPECTION		3RD SAMPLE INSPECTION	
Sample Size	Result	Action Required	Result	Action Required	Result	Action Required
A minimum of S Tubes per S.G.	C-1	None	N/A	N/A	N/A	N/A
	C-2	Plug or repair defective tubes and inspect additional 2S tubes in this S.G.	C-1	None	N/A	N/A
			C-2	Plug or repair defective tubes and inspect additional 4S tubes in this S.G.	C-1	None
			C-2	Plug or repair defective tubes and inspect additional 4S tubes in this S.G.	C-2	Plug or repair defective tubes
			C-3	Perform action for C-3 result of first sample	C-3	Perform action for C-3 result of first sample
			C-3	Perform action for C-3 result of first sample	N/A	N/A
	C-3	Inspect all tubes in this S.G., plug or repair defective tubes and inspect 2S tubes in each other S.G.  Notification to NRC pursuant to 10CFR50.73	All other S.G.s are C-1	None	N/A	N/A
			Some S.G.s C-2 but no additional S.G.s are C-3	Perform action for C-2 result of second sample	N/A	N/A
			Additional S.G. is C-3	Inspect all tubes in each S.G. and plug or repair defective tubes. Notification to NRC pursuant to 10CFR50.73	N/A	N/A

$S = 3 \frac{N}{n} \%$  Where N is the number of steam generators in the unit, and n is the number of steam generators inspected during an inspection.

NOTE: F\* tubes do not have to be plugged or repaired.

FARLEY-UNIT 2

3/4 4-15

AMENDMENT NO. 46, 63, 6

## REACTOR COOLANT SYSTEM

### BASES

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3. The tube plugging limit continues to apply to the portion of the tube in the entire upper joint region and in the lower roll expansion. As noted above the sleeve plugging limit applies to these areas also.
4. The tube plugging limit continues to apply to that portion of the tube above the top of the upper joint.

F\* tubes do not have to be plugged or repaired provided the remainder of the tube within the tubesheet that is above the F\* distance is not degraded. The F\* distance is equal to 1.79 inches and is measured down from the top of the tubesheet or the bottom of the roll transition, whichever is lower in elevation. Included in this distance is an allowance of 0.25 inch for eddy current elevation measurement uncertainty.

Steam generator tube inspections of operating plants have demonstrated the capability to reliably detect wastage type degradation that has penetrated 20% of the original tube wall thickness.

Whenever the results of any steam generator tubing inservice inspection fall into Category C-3, these results will be reported to the Commission pursuant to 10CFR50.73 prior to resumption of plant operation. Such cases will be considered by the Commission on a case-by-case basis and may result in a requirement for analysis, laboratory examinations, tests, additional eddy-current inspection, and revision of the Technical Specifications, if necessary.



UNITED STATES  
NUCLEAR REGULATORY COMMISSION  
WASHINGTON, D. C. 20555

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 64 TO OPERATING LICENSE NO. NPF-8

ALABAMA POWER COMPANY

JOSEPH M. FARLEY NUCLEAR PLANT, UNIT 2

DOCKET NO. 50-364

1.0 INTRODUCTION

By letter dated May 4, 1987, Alabama Power Company submitted a request for changes to the Joseph M. Farley Nuclear Plant, Unit 2, (Farley 2) Technical Specification (TS) 3/4.4.6, Steam Generators, surveillance and reporting requirements for testing tubes in the tubesheet region.

The amendment would allow operation with tube degradation in excess of the plugging or repair limits when the degradation is located below the F\* distance from the top of the tubesheet or the bottom of the roll expansion, whichever is lower. The F\* distance is defined as the length of continuous undergraded tube expansion in the tubesheet such that tube pullout would not occur during normal or postulated accident loading conditions. The bases for this change is described in Westinghouse Electric Corporation Report WCAP 11306, Revision 2, "Tubesheet Region Plugging Criterion for the Alabama Power Company Farley Nuclear Station Unit 2 Steam Generators," revised April 1987. The Westinghouse report was provided as Enclosure 1 in the May 4, 1987 letter.

The criteria for determining whether steam generator tube plugging and repair are required is described in Regulatory Guide (RG) 1.121, "Bases for Plugging Degraded PWR Steam Generator Tubes." This regulatory guide indicates three factors should be considered when establishing tube degradation limits. They are (1) the minimum tube wall thickness needed in order for tubes with defects to sustain the imposed loadings under normal operating conditions and postulated accident conditions, (2) an operational allowance for degradation between inspections, and (3) the crack size required to ensure the leakage per steam generator does not exceed the technical specifications limit. To determine the minimum wall thickness, the RG requires a margin of safety of not less than three against tube rupture under normal operating conditions and, the margin of safety required against tube failure under postulated accident conditions should be consistent with the margin of safety determined by the stress limits specified in NB-3225 of Section III of the ASME Code.

The Farley 2 steam generator tubes were fabricated with a full depth hardroll of the tubes within the tubesheet. When tubes are fabricated with a full depth hardroll, a transition region exists between the unrolled and fully expanded tube. In the definition of F\*, the bottom of the hardroll means the location where the tube has been fully expanded below

the transition region. In lieu of meeting wall thickness limitations, the F\* criterion establishes the length of fully expanded hardroll tube required to meet the wall thickness margins of safety required by RG 1.121.

## 2.0 DISCUSSION AND EVALUATION

Westinghouse Report WCAP-11306 indicates that the presence of the tubesheet will enhance the tube integrity in the region of the hardroll by precluding tube deformation beyond the initial expanded outside diameter. The resistance to both tubesheet rupture and tube collapse is strengthened by the presence of the tubesheet in that region. The result of the hardroll of the tube into the tubesheet is an interference fit between the tube and the tubesheet. This interference fit produces an elastic preload between the tube and the tubesheet. Tube rupture can not occur because the contact between the tube and tubesheet does not permit sufficient movement of the tube material. In a similar manner, the tubesheet does not permit sufficient movement of tube material to permit buckling collapse of the tube during postulated LOCA loadings. Thus, the only method of significantly damaging the pressure boundary provided by the tube is pullout of the tube from the tubesheet.

In hardrolled tubes, pullout is resisted by frictional force resulting from (1) the elastic preload between the tube and tube sheet, (2) thermal expansion, and (3) internal pressure. Frictional forces are reduced by tube sheet bowing. Westinghouse has laboratory tested steam generator tubing to determine the elastic preload caused by hardrolling. Forces resulting from thermal expansion, internal pressure and tube sheet bowing were determined analytically. Using a conservative coefficient of friction between the tube and tube sheet, Westinghouse has determined the net axial resistance to tube pullout for normal and faulted conditions. These net axial resistance forces were used to determine the hardrolled tube length required to meet the RG 1.121 safety margins against tube rupture during normal operation conditions and tube failure under postulated accident conditions. The F\* length includes (1) the distances required to meet RG 1.121 safety margins against tube rupture and failures, (2) a distance to account for eddy current measurement uncertainty and (3) a distance to account for reduced preload from material discontinuity at the hardroll transition and the postulated circumferential through-wall flaw. The Westinghouse analysis demonstrates that the F\* distance produces sufficient frictional forces to meet the margins of safety required by RG 1.121 and uncertainties in eddy current testing and material discontinuities.

In lieu of determining an operational allowance for degradation between inspections, the analysis postulated circumferential flaws propagated to a through-wall depth for 360 degrees around the tube. This results in a conservative estimation of flaw growth and precludes the need to determine operational allowances for degradation between inspections.

The F\* distance of roll expansion, which contains undegraded tubing, will prevent significant primary to secondary leakage from through-wall flaws

located below the F\* distance. The existing Technical Specification leakage rate requirements and accident analysis assumptions remain unchanged in the unlikely event that significant leakage from this region does occur. As noted above, the tube rupture and subsequent tube pullout is not expected for a tube using the F\* criteria. Any leakage out of the tube from within the tubesheet at any elevation in the tubesheet is fully bounded by the existing steam generator tube rupture analysis included in the Updated Final Safety Analysis Report. The proposed F\* criteria do not adversely impact any other previously evaluated design basis accident.

Based on the above, we conclude that the proposed changes to the steam generator tube plugging surveillance limits meet the safety margins and intent of RG 1.121. Also, the other related Technical Specification changes proposed by the licensee are acceptable.

### 3.0 ENVIRONMENTAL CONSIDERATION

This amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and changes to the surveillance requirements. The staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released off site, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that this amendment involves no significant hazards consideration, and there has been no public comment on such finding. Accordingly, this amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of this amendment.

### 4.0 CONCLUSION

The Commission made a proposed determination that this amendment involves no significant hazards consideration which was published in the Federal Register on June 17, 1987 at 52 FR 23093 on and consulted with the State of Alabama. No public comments were received, and the State of Alabama did not have any comments.

The staff has concluded, based on the considerations discussed above that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (2) such activities will be conducted in compliance with the Commission's regulations and the issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributor: B. Elliot

Dated:

Docket Nos.: 50-424  
and 50-425

SEP 22 1987

Mr. James P. O'Reilly  
Senior Vice President - Nuclear Operations  
Georgia Power Company  
P. O. Box 4545  
Atlanta, Georgia 30302

Dear Mr. O'Reilly:

Subject: Staff Acceptance of Vogtle Units 1 and 2 Offsite Dose  
Calculation Manual, Revision 4 (TACs 66076/66077)

By letter dated August 21, 1987, Georgia Power Company (GPC) submitted  
Revision 4 to the Vogtle Units 1 and 2 Offsite Dose Calculation Manual  
(ODCM). A page inadvertently omitted from the August 21 submittal was  
provided by letter dated September 17, 1987.

The staff has reviewed Revision 4 of the ODCM and has confirmed that it  
contains no technical changes from the previously approved Revision 3.  
The changes are in format and the correction of typographical errors.  
Therefore, Revision 4 of the ODCM is acceptable.

Sincerely,

ISI

Melanie A. Miller, Project Manager  
Project Directorate II-3  
Division of Reactor Projects, I/II

cc: See next page

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Vogtle Electric Generating Plant

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