UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

February 16, 2018

Mr. Robert S. Bement
Executive Vice President Nuclear/
Chief Nuclear Officer
Mail Station 7602
Arizona Public Service Company
P.O. Box 52034
Phoenix, AZ 85072-2034

SUBJECT: PALO VERDE NUCLEAR GENERATING STATION, UNIT 1 – RELIEF REQUEST NO. 57 TO APPROVE ALTERNATE REQUIREMENTS FOR THE REACTOR PRESSURE VESSEL HEAD NOZZLES TO PERFORM A BARE METAL EXAMINATION PER ASME CODE CASE N-729-4 (EPID L-2017-LLR-0132)

Dear Mr. Bement:

By letter dated October 26, 2017 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML17299B333), Arizona Public Service Company (the licensee), submitted Relief Request No. 57 (RR-57) to request relief from the requirements of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code Case N-729-4, "Alternative Examination Requirements for PWR [Pressurized Water Reactor] Reactor Vessel Upper Heads with Nozzles Having Pressure-Retaining Partial-Penetration Welds Section XI, Division 1," as conditioned by Title 10 of the *Code of Federal Regulations* (10 CFR) paragraph 50.55a(g)(6)(ii)(D), at Palo Verde Nuclear Generating Station (PVNGS), Unit 1.

Specifically, pursuant to 10 CFR 50.55a(z)(2), the licensee requested to use the proposed alternative in RR-57 on the basis that compliance with the specified ASME requirements would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety.

On November 1, 2017 (ADAMS Accession No. ML17305B236), the U.S. Nuclear Regulatory Commission (NRC) verbally authorized the use of RR-57 at PVNGS, Unit 1, during operating cycle 21 that ends in the spring of 2019. The enclosed safety evaluation describes the technical basis of the NRC's verbal authorization.

The NRC staff reviewed the licensee's submittal and determined that complying with the specified requirement would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety. Accordingly, the NRC staff concludes that the licensee has adequately addressed all of the regulatory requirements set forth in 10 CFR 50.55a(z)(2). Therefore, the NRC staff authorizes the use of RR-57 at PVNGS, Unit 1 for operating cycle 21.

All other ASME Code, Section XI, requirements for which relief was not specifically requested and approved remain applicable, including the third-party review by the Authorized Nuclear Inservice Inspector.

If you have any questions, please contact the Project Manager, Siva P. Lingam, at 301-415-1564 or via e-mail at Siva.Lingam@nrc.gov.

Sincerely,

Robert J. Pascarelli, Chief
Plant Licensing Branch IV
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. STN 50-528

Enclosure:
Safety Evaluation

cc: Listserv

UNITED STATES
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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELIEF REQUEST NO. 57 REGARDING ALTERNATIVE INSPECTION OF

REACTOR PRESSURE VESSEL CLOSURE HEAD PENETRATION NOZZLES

ARIZONA PUBLIC SERVICE COMPANY

PALO VERDE NUCLEAR GENERATING STATION, UNIT 1

DOCKET NO. 50-528

1.0 INTRODUCTION

By letter dated October 26, 2017 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML17299B333), Arizona Public Service Company (the licensee), submitted Relief Request No. 57 (RR-57) to request relief from the requirements of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (ASME Code) Case N-729-4, "Alternative Examination Requirements for PWR [Pressurized Water Reactor] Reactor Vessel Upper Heads with Nozzles Having Pressure-Retaining Partial-Penetration Welds Section XI, Division 1," as conditioned by Title 10 of the *Code of Federal Regulations* (10 CFR) paragraph 50.55a(g)(6)(ii)(D), "Augmented ISI [Inservice Inspection] requirements: Reactor vessel head inspections," at Palo Verde Nuclear Generating Station (PVNGS), Unit 1.

Specifically, pursuant to 10 CFR 50.55a(z)(2), "Hardship without a compensating increase in quality and safety," the licensee requested to use the proposed alternative in RR-57 on the basis that compliance with the specified ASME requirements would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety.

On November 1, 2017 (ADAMS Accession No. ML17305B236), the U.S. Nuclear Regulatory Commission (NRC) verbally authorized the use of RR-57 at PVNGS, Unit 1, during operating cycle 21. This safety evaluation describes the technical basis of the NRC's verbal authorization.

2.0 REGULATORY EVALUATION

Adherence to Section XI of the ASME Code is mandated by 10 CFR 50.55a(g)(4), "Inservice inspection standards requirement for operating plants," which states, in part, that ASME Code Class 1, 2, and 3 components (including supports) will meet the requirements, except the design and access provisions and the preservice examination requirements, set forth in Section XI of the ASME Code.

Pursuant to 10 CFR 50.55a(g)(6)(ii), "Augmented ISI program," the Commission may require the licensee to follow an augmented ISI program for systems and components for which the Commission deems that added assurance of structural reliability is necessary.

Enclosure

Paragraph 50.55a(g)(6)(ii)(D) of 10 CFR requires licensees of PWRs to augment their ISI of the reactor vessel head nozzles in accordance with ASME Code Case N-729 with conditions as a result of operating experience of primary water stress corrosion cracking (PWSCC) in reactor vessel head nozzles.

Section 50.55a(z), "Alternatives to codes and standards requirements," of 10 CFR states, in part, that "Alternatives to the requirements of paragraphs (b) through (h) of [10 CFR 50.55a] or portions thereof may be used when authorized by the Director, Office of Nuclear Reactor Regulation.... A proposed alternative must be submitted and authorized prior to implementation." The licensee must demonstrate that: (1) the proposed alternatives provide an acceptable level of quality and safety, or (2) compliance with the specified requirements would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety.

Based on the above, and subject to the following technical evaluation, the NRC staff finds that regulatory authority exists for the licensee to request the use of an alternative and the NRC to authorize the proposed alternative.

3.0 TECHNICAL EVALUATION

3.1 ASME Code Components Affected

Component:	Replacement Reactor Vessel Closure Head (RRVCH) Nozzles
Code Class:	Class 1
Examination Category:	ASME Code Case N-729-4
Code Item Number:	B4.40
Description:	Control Element Drive Mechanism (CEDM) Nozzles - Specifically, nozzles 22, 41, 46, 54, 55, 64, 73, 74, 81, 82, 86, 89, 90, and 97

There are 97 CEDM nozzles and 1 vent nozzle welded to the inside surface of the RRVCH with partial penetration J-groove welds.

3.2 Applicable Code Edition and Addenda

The third 10-year ISI interval edition for PVNGS, Unit 1, is the ASME Code, Section XI, 2001 Edition through 2003 Addenda. The examinations of the RRVCH penetrations nozzles are performed in accordance with ASME Code Case N-729-4, as conditioned by 10 CFR 50.55a(g)(6)(ii)(D).

The Manufacturing Code for the RRVCH in PVNGS, Unit 1, is the ASME Code, Section III, 1998 Edition through 2000 Addenda.

3.3 Applicable Code Requirements

Paragraph 3141, "General," of ASME Code Case N-729-4 states:

> (a) The VE [visual examinations] required by -2500 and performed in accordance with IWA-2200 and the additional requirements of this Case shall be evaluated by comparing the examination results with the acceptance standards specified in -3142.1.

(b) Acceptance of components for continued service shall be in accordance with -3142.

(c) Relevant conditions for the purposes of the VE shall include evidence of reactor coolant leakage, such as corrosion, boric acid deposits, and discoloration.

Paragraph 3142.1, "Acceptance by VE," of ASME Code Case N-729-4 states:

(a) A component whose VE confirms the absence of relevant conditions shall be acceptable for continued service.

(b) A component whose VE detects a relevant condition shall be unacceptable for continued service until the requirements of -3142.1(b)(1), (b)(2), and (c) below are met.

(1) Components with relevant conditions require further evaluation. This evaluation shall include determination of the source of the leakage and correction of the source of leakage in accordance with -3142.3.

(2) All relevant conditions shall be evaluated to determine the extent, if any, of degradation. The boric acid crystals and residue shall be removed to the extent necessary to allow adequate examinations and evaluation of degradation, and a subsequent VE of the previously obscured surfaces shall be performed, prior to return to service, and again in the subsequent refueling outage. Any degradation detected shall be evaluated to determine if any corrosion has impacted the structural integrity of the component. Corrosion that has reduced component wall thickness below design limits shall be resolved through repair/replacement activity in accordance with IWA-4000.

(c) A nozzle whose VE indicates relevant conditions indicative of possible nozzle leakage shall be unacceptable for continued service unless it meets the requirements of -3142.2 or -3142.3.

Paragraph 3142.2, "Acceptance by Supplemental Examination," of ASME Code Case N-729-4 states:

A nozzle with relevant conditions indicative of possible nozzle leakage shall be acceptable for continued service if the results of supplemental examinations [-3200(b)] meet the requirements of -3130.

Paragraph 3142.3, "Acceptance by Corrective Measures or Repair/Replacement Activity," of ASME Code Case N-729-4 states:

(a) A component with relevant conditions not indicative of possible nozzle leakage is acceptable for continued service if the source of the relevant

condition is corrected by a repair/replacement activity or by corrective measures necessary to preclude degradation.

(b) A component with relevant conditions indicative of possible nozzle leakage shall be acceptable for continued service if a repair/replacement activity corrects the defect in accordance with IWA-4000.

Paragraph 3200(b) of ASME Code Case N-729-4 states:

(b) The supplemental examination performed to satisfy -3142.2 shall include volumetric examination of the nozzle tube and surface examination of the partial-penetration weld, or surface examination of the nozzle tube inside surface, the partial penetration weld, and nozzle tube outside surface below the weld, in accordance with Fig. 2, or the alternative examination area or volume shall be analyzed to be acceptable in accordance with Mandatory Appendix I. The supplemental examinations shall be used to determine the extent of the unacceptable conditions and the need for corrective measures, analytical evaluation, or repair/replacement activity.

3.4 Reason for Request

During the Unit 1 refueling outage (1R20) at PVNGS, the RRVCH nozzle penetrations were examined in the as-found condition, using ASME Code Case N-729-4. During this examination, 14 penetrations were determined to have relevant conditions pursuant to ASME Code Case N-729-4, paragraph 3141(c).

As stated, in part, by the licensee in its letter dated October 26, 2017:

> Once it is determined that the possibility of nozzle leakage exists, [ASME] Code Case N-729-4, paragraph 3142.2, requires nozzles with relevant conditions indicative of possible nozzle leakage undergo supplemental examinations consisting of a volumetric examination of the nozzle tube and a corresponding surface examination in accordance with paragraph 3200(b). In order to perform the examinations in accordance with [ASME] Code Case N-729-4, paragraph 3200(b), it will be necessary to mobilize qualified personnel to the site on an emergent basis. It is estimated that mobilization of personnel and completion of the required supplemental examinations will take approximately five weeks depending upon availability of resources.
>
> In addition, the supplemental examination requires access to the underside of the highly contaminated RRVCH which would expose personnel to elevated dose rates. The additional dose is estimated to be approximately 0.75 to 1.0 person-rem [roentgen equivalent man] for this work.
>
> Adding this extra time duration to the outage and increasing the personnel dose for performing these supplemental examinations represents a hardship or unusual difficulty without a compensating increase in the level of quality and safety, pursuant to 10 CFR 50.55a(z)(2).

3.5 Licensee's Proposed Alternative and Basis for Use

3.5.1 Previous Examinations

As stated, in part, by the licensee in its letter dated October 26, 2017:

> The replacement partial-penetration welded nozzles of the Unit 1 RRVCH [at PVNGS], were examined prior to service. During fabrication, there were no indications detected by dye penetrant examination.... Ultrasonic examination (UT) and eddy current examination (ET) of 100% of the nozzles were performed following hydrostatic testing with no unacceptable indications.
>
> The Unit 1 RRVCH [at PVNGS], was installed in refueling outage 1R15 in May 2010.
>
> A VE was performed on the PVNGS RRVCH for Unit 1 in 2014 in accordance with ASME Code Case N-729-1, Table 1, item B4.40. The VE was performed by qualified examiners on the outer surface of the RRVCH, including the annulus area of the penetration nozzles. The VE did not reveal any relevant conditions that would be indicative of nozzle leakage.

3.5.2 MRP [Materials Reliability Program]-375 Information Regarding Structural Adequacy of the RRVCH

The licensee stated that evaluations were performed and documented in MRP-375, "Technical Basis for Reexamination Interval Extension for Alloy 690 PWR Reactor Vessel Top Head Penetration Nozzles," to demonstrate the acceptability of extending the inspection intervals for the components in ASME Code Case N-729-1, Item B4.40. Extended inspection intervals based on plant service experience, factor of improvement (FOI) studies using laboratory data, deterministic study results, and probabilistic study results are documented MRP-375. This information documents the structural suitability of the RRVCH for extended periods of time.

Much of the laboratory data in MRP-375 indicated an FOI of 100 for Alloy 690/52/152 versus Alloy 600/182/82 (for equivalent temperature and stress conditions) in terms of crack growth rates. In addition, laboratory and plant data demonstrate an FOI in excess of 20 in terms of the time to PWSCC initiation. This reduced susceptibility to PWSCC initiation and growth supports elimination of all volumetric examinations throughout the plant service period, and by extension, supports not performing volumetric examinations in 1R20.

As stated, in part, by the licensee in its letter dated October 26, 2017:

> Deterministic calculations demonstrate that the alternative volumetric re-examination schedule, [listed in MRP-375 of 15 years], is sufficient to detect any PWSCC before it could develop into a safety significant circumferential flaw that approaches the large size (i.e., more than 300 degrees of circumferential extent) necessary to produce a nozzle ejection. The deterministic calculations also demonstrate that any base metal PWSCC would likely be detected prior to a through-wall flaw occurring. Probabilistic calculations based on a Monte Carlo simulation model of the PWSCC process, including PWSCC initiation, crack growth, and flaw detection via ultrasonic testing, show a substantially reduced

effect on nuclear safety compared to a RRVCH with Alloy 600 nozzles examined per current requirements.

The licensee stated that the resistance of Alloys 690, 52 and 152 is demonstrated by the lack of PWSCC indications reported in these materials after more than 22 consecutive years of service. Several thick-wall and thin-wall applications are documented in MRP-375. This operating experience includes service at temperatures higher than those on the RRVCH and includes Alloy 690 wrought base metal and Alloy 52/152 weld metal. This experience includes ISI volumetric or surface examinations performed in accordance with ASME Code Case N-729-1 on 13 of the 41 RRVCH currently operating in the United States nuclear power plant fleet. This data supports a FOI in time to detectable PWSCC flaw initiation of at least five to 20 when compared to service experience of Alloy 600 in similar applications worldwide.

3.5.3 Recent RCS Operational Leakage Performance

From the operational data for PVNGS, Unit 1, cycle 20, the following information was submitted by the licensee to provide support for the conclusion that there is no evidence of an active nozzle leak:

- The surveillance test data from procedure 40ST-9RC02, *ERFDADS (Preferred) Calculation of RCS Water Inventory*, shows Reactor Coolant System (RCS) unidentified leakage rates were nominal.

- The containment atmosphere radiation monitor particulate channel RU-1 was nominal and constant.

- Containment tritium levels were nominal and constant.

- There was an increase in the containment east sump in-leakage levels which were attributed to secondary system leaks from the steam generator wet-layup pump seals.

- Chemistry samples taken of the residue in several locations on the RRVCH during 1R20 show that there is no evidence of short-lived radionuclides that must be present if there were a recent RCS leak.

- The ISI and boric acid program examinations, evaluations, and reports do not show evidence of a recent RCS leak.

The PVNGS, Unit 1, Technical Specification 3.4.14, "RCS Operational Leakage," requires monitoring of operational leakage and has the following limits:

a. No pressure boundary LEAKAGE;
b. 1 gpm [gallon per minute] unidentified LEAKAGE;
c. 10 gpm identified LEAKAGE; and
d. 150 gallons per day primary to secondary LEAKAGE through any one steam generator (SG).

This specification is applicable in operating Modes 1, 2, 3, and 4.

The licensee implements these requirements with station operating procedures; specifically, procedure 40ST-9RC02, *ERFDADS (Preferred) Calculation of RCS Water Inventory*, which is used to determine the leakage rates on a routine basis.

The licensee has the following RCS leak rate Action Levels to implement specific operational actions based on certain criteria or leakage trend data:

> Action Level 1 is reached when the rolling average of the last seven performances of the unidentified RCS leak rate exceeds 0.1 gpm, or the last nine consecutive unidentified RCS leak rates are greater than the baseline mean. Required actions are to document the event in a condition report, establish increased monitoring frequency of leakage indicators by daily performance of RCS water inventory performance, or as designated by the Shift Manager/Control Room Supervisor, and request assistance from the [RCS] engineer.
>
> Action Level 2 is reached when the last two consecutive unidentified RCS leakage rates are greater than 0.15 gpm, or two of three consecutive unidentified RCS leak rates are greater than the mean unidentified RCS leakage plus two standard deviations. Required actions are to ensure action level one response is completed, review recent plant evaluations to determine any suspect sources, evaluate changes in leakage detection indications, check any components or flowpaths whose condition has recently changed and check for recent maintenance activities.
>
> Action Level 3 is reached when the unidentified RCS leak rate is greater than 0.3 gpm, or greater than the mean plus three standard deviations. Required actions are to ensure action level one and two responses are completed, initiate planning for a containment entry per plant procedure 40DP-9ZZ01, *Containment Entry in MODE 1 thru MODE 4*, obtain a containment sump sample, chemistry to analyze the containment sump sample, evaluate other systems for indications of leakage, obtain a containment atmosphere sample for indications of RCS leakage, monitor containment sump east and west level, monitor reactor cavity sump level, monitor area radiation monitors, other containment parameters, identify the source of the leakage, determine the leakage rate, and initiate a plan to correct the leak.

Based upon the VE performed on the RRVCH and the operating time of the RRVCH, the licensee concluded that there is no leakage from any of the nozzles or partial penetration welds. The nozzle penetrations with relevant conditions were known to have been subjected to a prior spill emanating from the RRVCH vent valves, that radiochemistry results indicated no short half-life radioisotopes present and there was no evidence of carbon steel oxidation in the annulus of the nozzle penetrations with relevant conditions. Therefore, performing the stipulated emergent supplemental examinations of the nozzles would represent a hardship or unusual difficulty without a compensating increase in the level of quality and safety. The licensee proposes an alternative of performing a bare metal VE of the 14 applicable RRVCH nozzles within the scope of this relief request at the next refueling outage in accordance with ASME Code Case N-729-4.

3.6 Duration of Proposed Alternative

The licensee stated that the alternative is applicable until the end of operating cycle 21.

3.7 NRC Staff Evaluation

During the 1R20, the licensee performed a required VE of the RRVCH penetration nozzles. Fourteen nozzles were determined to have relevant conditions pursuant to ASME Code Case N-729-4. The licensee was unable to confirm that these relevant conditions were not indicative of possible nozzle leakage. In accordance with ASME Code Case N-729-4, paragraph 3142.2, nozzles with relevant conditions indicative of possible nozzle leakage must undergo supplemental inspections. The licensee identified a radiological dose hardship of 0.75 to 1 person-rem to perform these supplemental examinations over a possible 5-week period.

In lieu of the supplemental examinations required by paragraph 3142.2, the licensee proposed an alternative to perform a bare metal VE of the 14 applicable RRVCH nozzles at the next refueling outage in accordance with ASME Code Case N-729-4.

In order to support this proposed alternative the licensee noted the following:

- Previous examinations on the head found no indications;
- Structural adequacy of penetration nozzles and welds made with Alloy 690 materials;
- RCS operational leakage performance;
- Chemistry analysis of samples taken of the boric acid residue in 1R20; and
- Actions taken to review and inspect the 14 nozzles during the current refueling outage.

Due to these factors, the licensee stated that performing the required supplemental examinations of the 14 nozzles would represent a hardship or unusual difficulty without a compensating increase in the level of quality and safety.

The NRC staff reviewed the licensee's proposed alternative, technical basis, hardship and the as-found results of the RRVCH nozzles. The NRC staff finds that the licensee has provided sufficient information for the following NRC staff conclusions:

1. Given the operating experience, and use of Alloy 690 materials in the RRVCH, the licensee has demonstrated that the reactor vessel head degradation is not likely to occur in the next fuel cycle of operation.

2. The licensee's bare metal VE did not identify any areas of significant corrosion.

3. The licensee has demonstrated that there was an alternate possible source other than nozzle leakage of the relevant condition for each of the nozzles for which relief is requested.

4. The licensee's chemistry analysis supports the position that it is unlikely that an active leak exists.

The NRC staff notes that PWSCC, the active degradation mechanism, can challenge the structural integrity of the RRVCH in two ways. First, a PWSCC crack could grow circumferentially through the susceptible nozzle material to an extent, approximately 300 degrees, to cause a nozzle to be ejected. Second, leakage could cause significant boric acid corrosion of the low alloy steel head material causing a potential rupture in the reactor vessel head itself.

In order to address the issue of nozzle ejection, the licensee discussed the deterministic calculations provided in MRP-375 of the alternative re-examination schedule. The NRC staff finds that the likelihood of RRVCH nozzle ejection prior to the next refueling outage is unlikely. Therefore, the NRC staff finds that the licensee has provided sufficient basis to provide reasonable assurance of the structural integrity of the RRVCH in regards to nozzle ejection until the next refueling outage.

In order to address the issue of boric acid corrosion, the licensee could not rule out the possibility of leakage from a RRVCH penetration nozzle. Therefore, the potential of boric acid corrosion to begin over the next operating cycle could not be ruled out as well. However, the licensee did note that no significant corrosion was identified of the cleaned RRVCH during the recent refueling outage and that leakage rates of greater than 0.1 gpm would be required before boric acid corrosion of the upper head would challenge the structural integrity of the RRVCH. Enhanced leakage monitoring actions at this level of unidentified leakage will aid in providing reasonable assurance of structural integrity for the RRVCH. In order to address the leakage rate issue, the licensee included its Technical Specification requirements to monitor operational leakage and their RCS leak rate Action Levels, that will be implemented based on certain criteria or leakage trend data. The NRC staff reviewed each of the licensee's actions, and found in combination, they provided reasonable assurance of the structural integrity of the RRVCH for any potential leakage that could cause significant boric acid corrosion until the next refueling outage.

The NRC staff finds that performing the required supplemental examinations will add extra time to the outage and increase the personnel dose. The NRC staff finds that performing the required supplemental examinations, as compared to the licensee's proposed alternative, constitutes a hardship without a compensating increase in quality and safety.

Given the licensee's identified hardship, the NRC staff's review of the licensee's technical basis, and the enhanced leakage monitoring activities, the NRC staff finds that the licensee's proposed alternative to perform a bare metal VE of 14 reactor vessel head nozzles in accordance with ASME Code Case N-729-4, during the operating cycle 21, will provide reasonable assurance of the structural integrity of the RRVCH until the next scheduled VE.

4.0 CONCLUSION

As set forth above, the NRC staff determines that complying with the specified requirement would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety. Accordingly, the NRC staff concludes that the licensee has adequately addressed all of the regulatory requirements set forth in 10 CFR 50.55a(z)(2). Therefore, the NRC staff authorizes the use of RR-57 at PVNGS, Unit 1 for operating cycle 21.

All other ASME Code, Section XI, requirements for which relief was not specifically requested and approved remain applicable, including the third-party review by the Authorized Nuclear Inservice Inspector.

Principal Contributor: D. Render, NRR/DMLR/MPHB

Date: February 16, 2018

SUBJECT: PALO VERDE NUCLEAR GENERATING STATION, UNIT 1 – RELIEF REQUEST NO. 57 TO APPROVE ALTERNATE REQUIREMENTS FOR THE REACTOR PRESSURE VESSEL HEAD NOZZLES TO PERFORM A BARE METAL EXAMINATION PER ASME CODE CASE N-729-4 (EPID L-2017-LLR-0132) DATED FEBRUARY 16, 2018

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