TENNESSEE VALLEY AUTHORITY

Post Office Box 2000, Decatur, Alabama 35609-2000

May 13, 2026

10 CFR 21.21

ATTN: Document Control Desk
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555-0001

Browns Ferry Nuclear Plant, Unit 3
Renewed Facility Operating License No. DPR-68
NRC Docket No. 50-296

Subject: **Final Evaluation of 10 CFR 21 Notification -- Defect Associated with Anchor Darling Double-disc Gate Valve**

Reference:
1. "Wedge Pin Failure of an Achor/Darling Double-Disc Gate Valve at Browns Ferry Nuclear Plant Unit 1," dated February 25, 2013 (ML13064A012)
2. "Part 21 Wedge Pin Failure in Anchor Darling Motor Operated Double Disc Gate Valves with Threaded Stem to Upper Wedge Connections," dated March 1, 2013 (ML17194A825)
3. "Stem-Wedge Separation of an Anchor/Darling Double Disc Gate Valve at Exelon, LaSalle County Station, Unit 2, February 2017," dated July 11, 2017 (ML17199F890)
4. "Interim Report of a Deviation or Failure to Comply Associated with a Valve in the Unit 3 High Pressure Coolant Injection System," dated June 23, 2024 (ML24175A004)
5. "Updated Report of a Deviation or Failure to Comply Associated with a Valve in the Unit 3 High Pressure Coolant Injection System," dated August 22, 2024 (ML24235A497)
6. Letter from Flowserve to TVA, "10 CFR Part 21 Evaluation 114 Closure," dated October 28, 2024
7. "10 CFR 21 Notification -- Defect Associated with Anchor Darling Double-disc Gate Valve," dated July 10, 2025

The enclosed notification provides a revised evaluation of a defect associated with an Anchor Darling double-disc gate valve. The Tennessee Valley Authority (TVA) is

submitting this notification in accordance with Title 10 of the Code of Federal Regulations (10 CFR) 21.21(d)(3)(i), to submit information reasonably indicating a failure to comply or a defect affecting a basic component that is within his or her organization's responsibility and is supplied for a facility or an activity within the United States that is subject to the licensing, design certification, or approval requirements under 10 CFR 50.

On March 3, 2024, excessive valve stem rotation was observed while troubleshooting the Browns Ferry Nuclear Plant, Unit 3, High Pressure Coolant Injection (HPCI) outboard steam isolation valve. This issue was promptly entered into the BFN Corrective Action Program (CAP) as Condition Report (CR) 1914295.

TVA completed its 10 CFR 21 discovery process on April 24, 2024, and determined the need to perform a 10 CFR 21 Evaluation. This evaluation was logged in CAP as CR 1926691. The vendor, Flowserve, was contacted and assumed responsibility for performing the 10 CFR 21 Evaluation for this valve. In accordance with 10 CFR 21.21(b), Flowserve notified TVA on October 28, 2024, that they were not capable of performing the evaluation to determine if a defect exists, and that no definitive conclusions could be drawn with the available data.

Because of the inconclusive results, TVA procured additional engineering expertise to complete the required evaluation, and TVA recorded this in CAP as CR 1942523. An independent failure analysis was provided to Flowserve. This report concluded that the event was apparently caused by an improper upper wedge-to-stem joint, and the resulting mismatch in mating surface diameters resulted in the bending stress which led to the valve failure.

Following this report, a revised report was produced and approved by MPR on March 13, 2026, to incorporate additional input from the Flowserve Report and additional supporting scoping calculations. The report also responds to the discussion in Revision 1 of Flowserve's Report and includes updated recommendations. Lastly, the report incorporates additional EPRI reference documentation and broadens the recommended scope of the extent of condition review.

There are no new regulatory commitments contained in this letter. Should you have any questions concerning this submittal, please contact David J. Renn, Site Compliance Manager, at (256) 729-2636.

Respectfully,

Daniel A. Komm
Site Vice President

Enclosure: Final Evaluation of 10 CFR 21 Notification -- Defect Associated with Anchor Darling Double-disc Gate Valve

cc (w/ Enclosure):

NRC Regional Administrator - Region II
NRC Senior Resident Inspector - Browns Ferry Nuclear Plant
NRC Project Manager - Browns Ferry Nuclear Plant

ENCLOSURE

Browns Ferry Nuclear Plant
Unit 3

10 CFR 21 Notification
Defect Associated with an Anchor Darling Double-disc Gate Valve

See Enclosed

10 CFR 21 Notification
Defect Associated with Anchor Darling Double-disc Gate Valves

The following information is provided pursuant to 10 CFR 21.21(d)(4).

i. Name and address of the individual or individuals informing the Commission.

Daniel A. Komm
Site Vice President
TVA Browns Ferry Nuclear Plant
PO Box 2000
Decatur, AL 35609-2000

ii. Identification of the facility, the activity, or the basic component supplied for such facility or such activity within the United States which fails to comply or contains a defect.

Facility:	Browns Ferry Nuclear Plant (BFN), Unit 3
Docket No.:	50-296
License No.:	DPR-68
Basic Component:	A 10", Class 900 Anchor Darling double-disc gate valve (Vendor Drawing # W0025604; Serial # E125T-2-2) used as a High Pressure Coolant Injection (HPCI) outboard steam isolation valve in BFN, Unit 3 (3-FCV-073-0003).

iii. Identification of the firm constructing the facility or supplying the basic component which fails to comply or contains a defect.

The failed Anchor Darling double-disc gate valve was supplied by Flowserve.

Flowserve US, Inc.
1900 South Saunders Street
Raleigh, NC 27603

iv. Nature of the defect or failure to comply and the safety hazard which is created or could be created by such defect or failure to comply.

The deviation in this component was a stem fracture approximately two threads into the upper wedge assembly. 3-FCV-073-0003 is required for Primary Containment Isolation System (PCIS) function in the closed direction when in Modes 1, 2, and 3 and required to be open when in Mode 1 or in Modes 2 and 3 with reactor steam dome pressure > 150 lbs. to maintain HPCI Operability. As a result of the stem fracture, the valve was unable to perform its design function to open or close as required by TS.

v. The date on which the information of such defect or failure to comply was obtained.

The issue was entered into the BFN Corrective Action Program (CAP) on March 3, 2024, when the valve failure was discovered. On June 4, 2025, an evaluation was completed that determined that this condition represented a defect or failure to comply which is reportable in accordance with 10 CFR 21. The initial notification was submitted to the NRC Operations Center on July 10, 2025, and was recorded as Event Notification 57751. Interim reports regarding this issue were previously submitted on June 23, 2024; August 22, 2024; and November 27, 2024.

vi. In the case of a basic component which contains a defect or failure to comply, the number and location of these components in use at, supplied for, being supplied for, or may be supplied for, manufactured, or being manufactured for one or more facilities or activities subject to the regulations in this part.

All valves from Flowserve with threaded stem-to-upper wedge joint connections torqued to a higher value following the Flowserve Part 21, dated March 1, 2013, with 17-4 PH stem material operated at or above 500 degrees F. There are 18 valves at BFN that meet the initial conditions for the extent of condition. These valves will be further evaluated to determine if any can be excluded from the extent of condition.

vii. The corrective action which has been, is being, or will be taken; the name of the individual or organization responsible for the action; and the length of time that has been or will be taken to complete the action.

The corrective actions are as follows:

- Evaluate the Flowserve valves with threaded stem-to-upper wedge joint connections and 17-4 PH stem material operated at or above 500 degrees F to determine if the stems are susceptible to this failure. Replace any stem material determined to be susceptible to failure due to high torque and thermal embrittlement.
- Determine acceptable required torqued values (coefficient of friction) for the Flowserve valves with threaded stem-to-upper wedge joint connections with the original or replacement stem material.

viii. Any advice related to the defect or failure to comply about the facility, activity, or basic component that has been, is being, or will be given to purchasers or licensees.

Based on an independent technical review of this event, TVA recommends purchasers or licensees consider using a different valve design that eliminates this joint entirely or uses one that has more elongation in the stem threads which would minimize the potential for a loss of preload. Things to consider include the following:

- Thermal embrittlement of 17-4 PH is a known condition for operation in nuclear power plants but does not typically lead to failures on its own; therefore, it is proposed that the valve stem material should be changed from 17-4 PH to Inconel 718 or another material that is not susceptible to thermal embrittlement in

temperatures above 500 degrees F.

- It has been recommended to reduce the specific preload torque. The current method of preloading the joint by torquing the stem results in significant uncertainty because of large potential variation in friction in the threaded joint while being torqued. Achieving a consistent coefficient of friction is not a precise process, especially for a joint which employs a single fastener, like that of the stem-wedge connection. In addition, it is recommended that the axial load or deflection based preloading methods or load cell-based torque validation testing be considered as part of the preloading procedure to ensure that sufficient, but not excessive, preloads can be consistently achieved. Additionally, consider torquing and un-torquing the threaded joint multiple times to condition the threads, remove any local high spots, and reduce the variation in the axial preload resulting from a given torque.
- It is recommended that the installation and setup of a replacement valve stem assembly (i.e. disc-wedge pack fit-up) should be supervised by Flowserve to ensure the as-left dimensions and clearances are consistent with vendor's recommendations. A misalignment in the valve may lead to increased stress in the valve stem. Any potential misalignment should be prevented in future installations.
- An extent of condition review should be performed to identify other similar valves that may be susceptible to a similar failure. This review should be for all valves made by Flowserve that have the same upper wedge-to-stem joint connections and/or are susceptible to thermal embrittlement.

ix. In the case of an early site permit, the entities to whom an early site permit was transferred.

This event does not involve an early site permit.