TENNESSEE VALLEY AUTHORITY

Post Office Box 2000, Decatur, Alabama 35609-2000

July 10, 2025

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

The enclosed notification provides details 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 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.

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,

Quinn Leonard by delegation of authority

Daniel A. Komm
Site Vice President

Enclosure: 10 CFR 21 Notification -- Defect Associated with an 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

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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 and was submitted to the NRC Operations Center on June 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.

Six (6) valves of this size and type are used on BFN Units 1, 2, and 3 as HPCI inboard (BFN-1-FCV-073-0002, BFN-2-FCV-073-0002, BFN-3-FCV-073-0002) and HPCI outboard isolation valves on BFN Units 1, 2, and 3 (BFN-1-FCV-073-0003, BFN-2-FCV-073-0003, BFN-3-FCV-073-0003).

No valves of this type are currently in TVA's inventory. There are other valves of this design which may be subject to this defect in use at BFN and TVA Sequoyah Nuclear Plant. It is unknown if other valves of this design which may be subject to this defect are in use at other facilities operated by other utilities.

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.

This defect was documented in the BFN CAP as Condition Report 1914295. The current Corrective Actions (CAs) include:

- Coordinate with the vendor to determine a new valve stem material.
  - Owner: BFN Mechanical Design Engineering.
  - Projected completion date: August 6, 2025.
- Generate Work Orders to install stems made from the new material identified in the action above.
  - Owner: BFN Maintenance Program.
  - Projected completion date: September 5, 2025.
- Ensure that the vendor providing stem/wedge assemblies properly implements procedure/process revisions.
  - Owner: BFN Maintenance Program.
  - Projected completion date: September 19, 2025.

These CAs and their due dates may be revised, and additional CAs may be added as necessary as this situation unfolds. Multiple contributing factors have been identified, with a low likelihood of all these factors occurring in a single valve at once.

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.

Independent technical review of this event recommends that purchasers or licensees should 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. These reports additionally concluded that:

- Thermal embrittlement of 17-4 PH is a known condition for operation in nuclear power plants but does not typically lead to failures. The magnitude of thermal embrittlement changes based on the operating temperature and the material aging temperature (i.e., heat treatment). Since the valve's operating temperature cannot change, the magnitude of the stem's thermal embrittlement can be reduced by raising the aging temperature of the 17-4 PH material to H1100 or H1150. Changing the aging temperature will reduce the strength of the stem material, so the stem design should be reviewed to verify the acceptability of a lower material strength. If improvements to the wedge-to-stem joint are made such that proper bearing conditions occur and preloading the joint does not produce excessive bending stresses, the stem life is expected to vastly improve and be consistent with acceptable long-term performance of 17-4 PH material, as seen elsewhere in the plant and the industry.
- Utilities should perform extent of condition reviews. This review was not in the report's scope of work, but the report noted that other similar valves supplied by Flowserve may also contain the same improper upper wedge-to-stem joint. These extent of condition reviews should include all Flowserve double-disc valves with the same upper wedge-to-stem joint and a 17-4 PH stem. The following conditions could allow a utility to exclude a valve from these extent of condition reviews:
    - Valves operating at < 500°F, as this will likely prevent thermal embrittlement which would prevent the failure even if the improper wedge-to-stem joint is present.
    - Valves operated at > 500°F which have been in service for > 10 years and have been cycled > 30 times, because a valve with this defect would have already failed under those conditions.
- The valve's upper wedge-to-stem joint design possesses features which make it suboptimal for successful long-term operation. Specifically, the joint includes a short length of external stem threads that are threaded into internal threads in the upper wedge. When the joint is mated, applied torque on the stem results in an axial preload in the stem that is reacted in bearing between the stem collar and the upper wedge. Since the length of the threads is short, the amount of elongation in the stem is very small. Any condition or load that interferes with, reduces, or degrades the bearing between the stem collar and upper wedge will cause a decrease in the elongation and loss or reduction of the preload. If the preload is lost such that friction between the stem collar and upper wedge at the bearing joint cannot carry the applied torque to the stem, this torque will be applied to the pinned joint between the stem and upper wedge. Furthermore, if the preload is lost entirely, the portion of the stem below the collar can be subjected to loads which it was unintended to carry. While it was concluded that the improper wedge-to-stem joint caused a reduction in the preload applied during the initial torquing, and that this contributed to the failure, it is unclear if the preload was further reduced or lost during operation. Utilities

should 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.

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.