ENCLOSURE 2

M240075

Final Report Notification Information per 10 CFR 21.21(d) and Safety Communication

Non-Proprietary Information

IMPORTANT NOTICE

This is a non-proprietary version of Enclosure 1 of M240075, which has the proprietary information removed. Portions of the document that have been removed are indicated by an open and closed bracket as shown here [[ ]].

10 CFR Part 21 Communication

Global Nuclear Fuel

SC *24- 01* R0 ***April 22, 2024***
008N5282 Revision 0

To: ***Constellation Energy Group – See Attachment 1***

Subject: ***GNF3 Spacer Relocation***

X	**Reportable Condition [21.21(d)]**		**60 Day Interim Report [21.21(a)(2)]**
	Transfer of Information [21.21(b)]	X	**Safety Information Communication**

Summary:

During the cycle 20 refueling outage and inspection of GNF3 fuel assemblies at LaSalle County Station Unit 1 in February 2024, a bundle with five (5) relocated upper spacers was discovered. Later, during the cycle 20 refueling outage at Limerick Unit 1 in April 2024, a similar condition, four (4) relocated upper spacers, was also discovered. The unlatched (incorrectly assembled) water rod condition was previously reported in 2022 and the driver for the inspections at LaSalle and Limerick. However, the relocated spacers found at these plants are a new discovery and not predicted by prior evaluations. The LaSalle and Limerick plants discovered the condition during shutdown. This communication documents the extent of the condition for operating plants with susceptible bundles, which includes the FitzPatrick and Peach Bottom Unit 2 plants.

The results of GNF's evaluations conclude that the condition constitutes a reportable condition. The safety functions for a basic component, such as a GNF3 fuel bundle, as defined in 10 CFR 21.3 are not compromised to the extent that there is a major reduction in the degree of protection provided to public health and safety. However, relocated spacers in a GNF3 fuel assembly could result in degraded critical power margin. The condition, should it exist in fuel being operated in either Fitzpatrick or Peach Bottom Unit 2 could contribute to exceeding the Technical Specification Safety Limit Minimum Critical Power Ratio values specified for these plants. Please contact me if there are any questions.

Issued by: Michelle P. Catts

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Background

In references 1 and 2, in November 2022, GNF reported the discovery of a raised water rod in a GNF3 fuel bundle. This bundle was discovered at Grand Gulf Nuclear Station (GGNS) during refueling inspection and the subsequent investigation determined that the water rod (WR) was not correctly installed in the fuel assembly. Incorrect installation can allow the WR to move under the influence of hydraulic forces. Similarly, this condition could allow movement of spacers, although GNF's initial evaluation did not predict that this would happen during the first cycle of operation. On February 21, 2024, GNF learned that during the refueling outage for LaSalle County Station Unit 1, a GNF3 fuel bundle was discovered with five (5) spacers collocated below the upper tie-plate (UTP). Initially, three (3) mislocated spacers were observed by camera. Later, with different lighting and a different camera angle, it was discovered that five (5) spacers (not 3) had relocated to the top of the bundle. When spacers are in their correct positions, they are well below the UTP and difficult to see from the top of the bundle, especially under conditions in-reactor or in a refueling pool. This bundle, designated GGJ319, was the only bundle at LaSalle 1 identified with this condition.

During the course of the investigation of the relocated spacers at LaSalle Unit 1, the Limerick Unit 1 nuclear plant entered its cycle 20 refueling outage and began WR inspections. On April 12, 2024, during these inspections, a second single instance of relocated spacers was discovered. The discovery at Limerick 1 is more recent than LaSalle 1, so evaluations of the bundle are ongoing. However, visual inspections confirm that the upper four (4) spacers were collocated below the UTP, and the lower four (4) spacers remained in their correct positions. The observed condition was limited to a single bundle, designated GGK123.

Discussion

Spacer movement is normally prohibited within a fuel bundle by design. In a correctly manufactured fuel assembly, spacers are secured and locked into place by tabs on the WR. The WR tabs provide a "mechanical stop" above and below the spacers and prevent vertical movement (when the WR is properly installed and oriented during bundle assembly). The apparent cause for the relocated spacers is that the WR was not properly installed in GNF's factory. Specifically, the WR was not properly engaged, rotated and locked into place, which would allow unconstrained movement of both the WR and spacers. The WR as an underlying cause is consistent with the discovery at GGNS.

The remaining plants (post LaSalle) with potentially susceptible GNF3 bundles are summarized in Table 1. Since GNF issued its safety communication for this issue in 2022, potentially affected plants have been performing inspections during refueling outages. The table is based on reference 2 data and has been updated for recent inspection results. After the discovery at LaSalle, but prior to issuance of this report, some of the potentially affected plants entered shutdown (including Limerick). In the table, the term "shutdown" means that the plant has entered or will enter a refueling outage.

Table 1 Post LaSalle plants with susceptible GNF3 bundles and their status

Plant	Status	Number of Suspect Bundles
Grand Gulf Nuclear Station	• Shutdown • WRs inspected • No defects found	307
Nine Mile Point Unit 2	• Shutdown • WRs inspected • No defects found	308
Fermi	• Shutdown • WRs inspected • No defects found	12
Limerick Unit 1	• Shutdown • WRs inspected • One (1) defect found	268
Peach Bottom Unit 2	Shutdown expected Fall 2024	156
FitzPatrick	Shutdown expected Fall 2024	140

After the discovery of relocated spacers at LaSalle, GNF initiated an internal condition report as required by quality procedures. As part of the GNF corrective action process, a root cause analysis (RCA) was initiated. The RCA is ongoing and will be completed subsequent to this report. This evaluation of potential effects and extents must be completed without the additional insights that could be gained from RCA results.

The assessment discussed in this report assumes that the conditions observed at LaSalle and Limerick could exist at an operating plant in the population with uninspected first cycle GNF3 bundles ("susceptible bundles"). The physical mechanisms responsible for movement (hydraulic forces versus frictional resistance) are inferred or assumed, but their relative magnitudes may depend on a number of variables that are unknown in a particular plant. Also, at this point, it is unknown if other contributing factors exist. Therefore, this evaluation assumes that all spacers in suspect bundles can move, even though the lower spacers at LaSalle and Limerick remained in place. In general, the more conservative bundle configuration (with respect to the spacer arrangement) is considered in all the GNF analyses for this condition, without regard to possible causes.

No instances of multiple, improperly assembled GNF3 bundles in a single plant and cycle have been previously observed. There is no evidence to suggest that this is impossible, but observations and the very low probability of uninstalled WRs (discussed in PRC 22-04) indicate multiple impacted assemblies at the same site is highly improbable. Therefore, the assumed condition in this evaluation is based on a single instance of an improperly assembled GNF3 bundle in a plant operating with susceptible bundles. Potentially affected bundles are assumed to be independent and infrequent, so that multiple instances in a single plant / cycle are much less probable than a single instance.

Based on inspection results as of April 12, 2024, a total of 2,510 of 2,806 bundles from the suspect population have been inspected. Out of 2,510 bundles, four (4) were determined to have improperly installed WRs, including LaSalle and Limerick. Therefore, 4 / 2,510 = 0.0016 or a 0.16% defect rate has been observed, which is consistent with values reported for human error probabilities (reference 3) for skilled workers (~0.01 to ~0.001), the original estimate (reference 1), and the assumption that two or more affected bundles in a core is unlikely. Of the four "observed" affected bundles, only two instances of relocated spacers have actually been witnessed[1]. If the 0.16% value is taken as an estimate of the defect rate, then based on the Table 1 suspect bundle populations, less than one (0.0016 x 296 < 1) affected bundle is expected in the remaining population.

Probabilities of occurrence are shown in Table 2. The statistics assume a binomial distribution. For a binomial distribution: (1) two possible outcomes, pass or fail, defective or not defective, are possible for each suspect bundle, (2) each suspect bundle is an independent "trial" (inspection result or possibility of existence), and (3) the pass-fail probability is the same (constant) for each of the trials. In the table, "n" is the number of suspect bundles (i.e., a trial) in a plant, or the total for both plants, "x" is the number of defective bundles, and $p = 1.60 \times 10^{-3}$ is the probability of a defective bundle. The binomial probabilities f(x|n,p) and cumulative probabilities F(x|n,p) are provided in the table. Using the historical defect rate, the probability "x" defects is f(x|n,p) and the probability of "≤ x" defects (i.e., x_j, x_{j-1}, x_{j-2}, … , 0 defects) is F(x|n,p).

Table 2 Binomial probabilities for the remaining operating plants

x = # defective	**n = 296 (Total)**		**n = 156 (Peach Bottom 2)**		**n = 140 (FitzPatrick)**	
	f(x|n,p)	F(x|n,p)	f(x|n,p)	F(x|n,p)	f(x|n,p)	F(x|n,p)
x = 0	0.624	0.624	0.780	0.780	0.800	0.800
x = 1	0.295	0.918	0.194	0.974	0.179	0.979
x = 2	0.069	0.988	0.024	0.998	0.020	0.998
x = 3	0.011	0.999	0.002	1.000	0.001	1.000

Using the observed historical defect rate, the probability of observing zero defective bundles in the remaining 296 suspect bundles is 0.624. Similarly, the probability of observing one defective bundle is 0.295 and the probability of observing two defective bundles is 0.069. One or zero defectives are the two most likely outcomes (x ≤ 1 with 91.8% confidence, i.e., F(x|n,p) = 0.918 for n = 296). Two or three defective bundles are statistically unlikely based on the historical rate.

[1] The discovery at Clinton station was deemed a defect. The WR did not appear to be properly rotated. The spacer locations were not checked, so spacer movement is unknown and not ruled out. Also, the discovery at Grand Gulf Nuclear Station was relatively early, before significant relaxation of spring forces could occur, which would permit movement. So, it is not known if spacer relocation would have occurred later at GGNS.

Evaluation

Spacer Functions

Within a fuel bundle, spacers have defined design functions or can influence fuel assembly functions. The relevant design functions are summarized below. Safety functions can be compared against the design functions to evaluate the potential impact of relocated spacers.

- Mechanical – "Spacers," as their name implies, maintain rod-to-rod spacing. Spacers provide radial positioning within the assembly and contribute to the overall mechanical stiffness (rigidity) of the bundle.
- Nuclear – The spacer's positioning function is a mechanical design feature that provides an optimal nuclear design (i.e., the moderator-to-fuel ratio within the fuel lattice).
- Thermal-hydraulic – Spacers influence the overall hydraulic resistance (pressure drop) of the fuel bundle, which is considered in steady-state, transient (including stability), and design basis accident evaluations. Spacers are a design feature that prevents determinantal flow induced vibration and promotes efficient cooling.

The potential impacts on safety and plant technical specifications are evaluated considering the spacer functions identified above.

Safety Function Evaluation

Safety functions of basic components are defined as those necessary to assure:

A. The integrity of the reactor coolant pressure boundary;
B. The capability to shut down the reactor and maintain it in a safe shutdown condition; or
C. The capability to prevent or mitigate the consequences of accidents which could result in potential offsite exposures comparable to those referred to in § 50.34(a)(1), § 50.67(b)(2), or § 100.11 of 10 CFR, as applicable.

The potential impact of relocated spacers have been evaluated for each item, A – C above, including shutdown margin (SDM), bundle seismic performance, and accident scenarios, e.g., loss of coolant accident (LOCA) evaluations. Potentially relocated spacers do not have a significant impact on these functions.

Evaluation of Technical Specification Safety Limits

Under 10 CFR Part 21.3, a defect includes: "(4) A condition or circumstance involving a basic component that could contribute to the exceeding of a safety limit, as defined in the technical specifications of a license for operation issued under part 50 or part 52 of this chapter." In general, for a BWR/4, safety limits apply to the reactor core and reactor coolant system pressure (i.e., dome pressure). Reactor core safety limits include:

a. Limits on core thermal power
b. Minimum critical power ratio (MCPR), usually termed the Safety Limit MCPR (SLMCPR)
c. Reactor vessel water level

The condition observed at LaSalle and Limerick will not significantly impact total core thermal power or reactor water level. However, potentially relocated spacers could impact MCPR margin to the Technical Specification SLMCPR (item "b" above).

For LaSalle, the potential impact of a raised WR and relocated spacers was evaluated. The focus of the prior LaSalle analysis was steady-state operation because it was known that a transient or accident event did not occur. The situation is similar for the more recent discovery at Limerick. This evaluation addresses both steady-state and transients for the susceptible plants shown in Table 1, namely Peach Bottom Unit 2 and FitzPatrick.

Steady-State MCPR Margin

In general, for GNF fuel products, critical power margin is evaluated using the GE critical quality (X) versus boiling length (GEXL) correlation. For GNF3, the specific version of this correlation is GEXL21, which is based on full-scale steam-water data. The underlying data was obtained from electrically heated replicas of the GNF3 assembly, with intact (i.e., properly located) spacers. In order to compute a "delta" or adjustment factor for application of GEXL21 to a prediction for a bundle with relocated spacers, GNF has utilized data [[

]] This multiplier can be further adjusted for biases in the scaling method, resulting in a [[ ]] "penalty" factor (i.e., a multiplier on the calculated GEXL21 critical power) for a GNF3 bundle with potentially relocated spacers.

If the condition exists in an operating plant, critical power performance in a potentially affected bundle will be degraded and there will be potential to exceed the SLMCPR. The steady state MCPR margin necessary to assure that the operating MCPR remains above the safety limit would be MCPR > 1.07 / [[ ]]. Note, however, that GNF has performed statistical evaluations that demonstrate that 99.9% of the fuel rods in potentially affected cores will avoid steady-state boiling transition at MCPR values lower than [[ ]].

Transient MCPR Margin

General design criterion GDC 10 from 10 CFR Part 50 Appendix A, reactor design, states that "the reactor core and associated coolant, control, and protection systems shall be designed with appropriate margin to assure that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including the effects of AOOs." In the GNF thermal analysis methodology, for the MCPR thermal limit, this is accomplished by setting the OLMCPR margin so that it protects the Technical Specification SLMCPR. Therefore, in general, the OLMCPR provides MCPR margin for the AOO events described in a BWR plant's UFSAR.

GNF's single rod tests were used to evaluate the impact of relocated spacers on the GNF3 critical power. Similarly, single rod transient test data has been used to evaluate the impact of "no spacers" on GNF3 transient ΔCPR/ICPR. Based on these comparisons there is no evidence of any significant effect of spacers on the transient ΔCPR/ICPR. The biases for the tests are small compared to uncertainties. In other words, if transient margin calculations are performed to assess this condition, they do not need to consider additional "penalty" factors or adjustments for dynamic behavior.

Conclusions

GNF has performed an evaluation for potentially affected plants. The conclusions are as follows:

- The condition observed at LaSalle, i.e., relocated spacers, is a new discovery and not previously evaluated in GNF safety communications. There are two potentially affected plants. Specifically, the condition could exist at either or both FitzPatrick and Peach Bottom Unit 2. These two plants are currently operating with GNF3 bundles in the suspect population identified by References 1 and 2.
    - The condition is possible, but not known to exist with certainty. GNF has determined that a low but non-zero possibility exists based on statistical and causal analyses and the dates of manufacture of certain GNF3 fuel products. For the 296 remaining suspect bundles, the probability of finding zero (0) or one (1) affected assemblies is 0.624 and 0.295, respectively. The probability of finding two (2) or more affected bundles in one plant is 1 – 0.998 = 0.002.
- The potential condition of a raised WR and relocated spacers is a defect that can result in reduced MCPR margin. Considerations for continued operation with susceptible bundles include:
    - Under steady-state conditions, appropriate MCPR margin should be applied to potentially affected GNF3 bundles to assure adequate protection.
    - If this condition exists and the plant experiences a transient event, such as a limiting or near limiting AOO, unless additional margin is applied, it is expected that the SLMCPR may be temporarily exceeded for a short duration. It is also possible that limited fuel damage could occur, i.e., some number of rods within a potentially affected bundle could experience heat up and cladding could fail.

"Limited damage" means that this condition only applies to the potentially affected bundle. Rod-by-rod susceptibility within that bundle depends on a number of factors. Factors that generally influence rod susceptibility to boiling transition include the rod specific local power, type of rod, and position within the lattice, although this last item is expected to be less significant for a bundle with relocated spacers. In general, full length, higher power rods are more susceptible to boiling transition than partial length and/or lower power rods.

The potential existence of a defect in a basic component (i.e., unlatched WRs in GNF3 fuel) has been previously reported (references 1 and 2). However, the relocated spacers at LaSalle 1 and Limerick 1, which are a potential consequence of an unlatched WR is a new discovery. Neither of these plants experienced fuel damage as a result of the condition. GNF is making this report because of the potential effect on margin to the TS MCPR limit. The condition, should it exist in fuel being operated in either Fitzpatrick or Peach Bottom Unit 2, could contribute to exceeding the Technical Specification Safety Limit Minimum Critical Power Ratio values specified for these plants, as well as limited fuel damage. Based on NEI 14-09 guidance and the potential for exceeding the TS MCPR, the condition is reportable.

Recommended Actions

GNF recommends that potentially affected licensees consider the following actions:

- Review the GNF evaluation as input to any evaluations performed to satisfy reporting requirements applicable to historical operating conditions.
- Review the GNF evaluation as input to support any required evaluations for fuel and continued operation.

References

1. Letter M220152, Halac to US NRC, Subject: Part 21 60-Day Final Report Notification: GNF3 Raised Water Rod (SC-22-01 Revision 2), November 30, 2022.
2. Letter M220152, Enclosure 2, 10 CFR Part 21 Communication, GNF3 Raised Water Rod – Final Notification, November 30, 2022.
3. NUREG/CR-1278, Handbook of Human Reliability Analysis with Emphasis on Nuclear Power Plant Applications, August 1983.
4. 006N2027, Supplemental Reload Licensing Report for Peach Bottom Unit 2 Reload 24 Cycle 25, Revision 1, October 2022.
5. 006N2045, Supplemental Reload Licensing Report for FitzPatrick Reload 25 Cycle 26, Revision 1, September 2022.

Acronyms

Acronym	Definition
AOO	Anticipated Operational Occurrence
BWR	Boiling Water Reactor
CPR or MCPR	(Minimum) Critical Power Ratio
GEXL	GE critical quality (X) versus boiling Length
LOCA	Loss of Coolant Accident
MFLCPR	Maximum Fraction of Limiting CPR
OLMCPR	Operating Limit MCPR
RC	Reportable Condition
RCA	Root Cause Analysis
RPV	Reactor Pressure Vessel
SC	Safety Communication
SDM	Shutdown Margin
SLMCPR	Safety Limit MCPR
TS	Technical Specification
UFSAR	Updated Final Safety Analysis Report
UTP	Upper Tie-Plate
WR	Water Rod

Attachment 1
List of Potentially Affected Plants

BWR Plants and Associated Facilities

	Utility	Plant
_____	Constellation	Nine Mile Point 1-2
_____	Constellation	Clinton
_____	Constellation	Dresden 2-3
X	Constellation	FitzPatrick
_____	Constellation	LaSalle 1-2
_____	Constellation	Limerick 1-2
X	Constellation	Peach Bottom 2-3
_____	Constellation	Quad Cities 1-2
_____	Detroit Edison Co.	Fermi 2
_____	Duke Energy - Progress	Brunswick 1-2
_____	Energy Northwest	Columbia
_____	Entergy	Grand Gulf
_____	Entergy	River Bend
_____	Energy Harbor	Perry 1
_____	Nebraska Public Power District	Cooper
_____	Talen Energy	Susquehanna 1-2
_____	Public Service Enterprise Group Incorporated	Hope Creek
_____	Southern Nuclear Operating Co.	Hatch 1 - 2
_____	Southern Nuclear Operating Co.	Pooled Equipment Inventory Co.
_____	Tennessee Valley Authority	Browns Ferry 1-3
_____	Xcel Energy	Monticello

Domestic Sodium Fast Reactor Plants

	Utility	Plant
_____	PacifiCorp	Kemmerer 1

US PWR Plants

	Utility	**Plant**
_____	AmerenUE	Callaway
_____	Arizona Public Service	Palo Verde 1-3
_____	Constellation	Braidwood 1-2
_____	Constellation	Byron 1-2
_____	Constellation	Calvert Cliffs 1-2
_____	Constellation	Fort Calhoun
_____	Constellation	Ginna
_____	Constellation	Three Mile Island 1
_____	Dominion	Millstone 3
_____	Dominion	North Anna 1-2
_____	Dominion	Surry 1-2
_____	Duke Energy Corporation	Catawba 1-2
_____	Duke Energy Corporation	Crystal River 3
_____	Duke Energy Corporation	McGuire 1-2
_____	Duke Energy Corporation	Oconee 1-3
_____	Duke Energy Corporation	Robinson
_____	Duke Energy Corporation	Shearon Harris
_____	FirstEnergy Nuclear Operations Co.	Beaver Valley 1-2
_____	FirstEnergy Nuclear Operating Co.	Davis-Besse
_____	Florida Power & Light	Seabrook
_____	Florida Power & Light	St. Lucie 1-2
_____	Florida Power & Light	Turkey Point 3-4
_____	Florida Power & Light	Point Beach 1-2
_____	Indiana Michigan Power Corp	D C Cook 1-2
_____	Northern States Power	Prairie Island 1-2
_____	Pacific Gas & Electric Co.	Diablo Canyon 1-2
_____	PSEG Nuclear LLC	Salem 1
_____	PSEG Nuclear LLC	Salem 2
_____	South Carolina Electric & Gas Co.	Summer
_____	South Texas Project Nuclear Operating Co.	South Texas Project 1-2
_____	Southern California Edison Co.	San Onofre 2-3
_____	Southern Nuclear Operating Co.	Farley 1-2
_____	Southern Nuclear Operating Co.	Vogtle 1-2
_____	Tennessee Valley Authority	Sequoyah 1-2
_____	Tennessee Valley Authority	Watts Bar 1
_____	Tennessee Valley Authority	Watts Bar 2
_____	TXU Electric Generation Co.	Comanche Peak 1-2
_____	Wolf Creek Nuclear Operating Corp.	Wolf Creek

Attachment 2 – Recent GE Hitachi Nuclear Energy 10 CFR Part 21 Communications

The following is a list of recent 10 CFR Part 21 communications that GE Hitachi Nuclear Energy (GEH) and Global Nuclear Fuel (GNF) have provided to affected licensees as Reportable Conditions (RC), Transfers of Information (TI), 60-Day Interim Reports (60 Day) and/or Safety Information Communications (SC).

Number	Ref.	Subject	Date
SC 24-01	PRC 24-01	GNF3 Spacer Relocation	04/22/2024

Attachment 3
GNF's Assessment of 06.07 Scope of Operability Determinations

Item	Point to Address	GNF Assessment
a.	Possible elements of an OD include:	
(1)	The SSC affected by the condition,	The condition is limited to GNF3 fuel assemblies within specific dates of manufacture. Reference: Letter M220152, Halac to US NRC, Subject: Part 21 60-Day Final Report Notification: GNF3 Raised Water Rod (SC-22-01 Revision 2), November 30, 2022. In subsequent entries, susceptible GNF3 fuel is referred to as "potentially affected," because the condition is not known to exist with certainty.
(2)	The extent of condition for all similarly affected SSCs,	The condition is that potentially affected fuel assemblies with unrotated water rods (refer to item #1) may allow spacers to shift and relocate to the upper tie-plate. This condition may affect critical power margin. There are no other affected SSCs.
(3)	The CLB requirements or commitments established for the affected SSC,	The current licensing basis (CLB) requirement of concern for the potentially affected GNF3 fuel is adherence to the Technical Specification MCPR limit.
(4)	The specified safety function(s) performed by the affected SSCs,	GNF3 fuel, must retain adequate structural integrity to ensure that the core can be shut down and maintained in a safe condition, as well as minimize the consequences of accidents. These safety functions have been evaluated by GNF and no concerns were identified. Potentially affected GNF3 fuel must be operated above the Technical Specification (TS) Safety Limit Minium Critical Power Ratio (SLMCPR). GNF's evaluation identified a potential effect on MCPR margin.

Item	Point to Address	GNF Assessment
(5)	The effect or potential effect of the condition on the affected SSC's ability to perform its specified safety function(s), and	The potential effect of the condition does not affect GNF3 fuel's ability to perform its specified safety functions (e.g., achieve and maintain safe shutdown). However, the condition, if it exists, impacts a potentially affected GNF3 bundle's MCPR margin (i.e., margin to the Technical Specification MCPR limit).
(6)	Whether there is a reasonable assurance of operability, including the basis for the determination and any compensatory measures put in place to establish or restore operability.	A reasonable assurance of operability can be established by analysis. GNF's analysis and basis is available to the affected licensee.
b.	The following things should be considered when reviewing ODs:	
(1)	Design basis events are plant-specific, and plant-specific TS, bases, and safety evaluations may contain plant-specific considerations related to operability,	GNF's evaluation of operating MCPR margin considers TS bases, GNF3 design bases, and plant specific bases (e.g., core designs for the potentially affected plant's current operating cycles).
(2)	An SSC's operability requirements are based on safety analyses of specific design basis events for one mode or specified condition of operation and may not be the same for other modes or conditions of operation; therefore, all applicable modes and conditions of operation should be considered,	GNF's evaluation is considered as "input" or support for the affected licensee's evaluation. The GNF evaluation considers GNF3 design bases and appropriate conditions of plant operation to support the licensee's operability determination.
(3)	The operability requirements for an SSC encompass all necessary support systems (per the TS definition of operability) regardless of whether the TS explicitly specifies operability requirements for the support functions,	GNF's evaluation considers GNF3 fuel as part of the "core system," so, interactions and other system interdependencies are adequately addressed for operation.

Item	Point to Address	GNF Assessment
(4)	In order to evaluate conditions, it is assumed in the OD that the design basis event occurs. The occurrence of multiple simultaneous design basis events should be considered only to the extent that they are required as a part of the plant's CLB, and	GNF's evaluation considers design basis events for GNF3 fuel that are appropriate for the CLBs of potentially affected plants.
(5)	Compensatory measures may be established to restore or maintain operability of an SSC. See section 06.08 of this IMC for additional guidance on compensatory measures.	GNF's assessment of compensatory measures is that application of normal core and plant surveillance systems and processes will be adequate to monitor and ensure continued operability of GNF3 fuel.