10165

Northeast
Utilities System

107 Selden Street, Berlin, CT 06037

Northeast Utilities Service Company
P.O. Box 270
Hartford, CT 06141-0270
(203) 665-5000

DS09

1994 APR -9 PM 3:10

April 5, 1994

Docket Nos. 50-213
50-245
50-336
50-423
B14807

Mr. David L. Meyer, Chief
Rules Review and Directives Branch
Division of Freedom of Information
and Publication Services
Office of Administration
U.S. Nuclear Regulatory Commission
Washington, DC 20555

D. Allison
59FR5614
2/7/94
(9)

Dear Mr. Meyer:

Haddam Neck Plant
Millstone Nuclear Power Station, Unit Nos. 1, 2, and 3
Second Draft NUREG-1022, Revision 1 Comments

On February 7, 1994, the NRC issued for public comment a draft of Revision 1 to NUREG-1022, "Event Reporting Guidelines, 10CFR50.72 and 50.73, Second Draft Report for Comment." The NRC Staff developed this document to clarify existing guidance related to the reporting of potentially safety significant events and conditions pursuant to 10CFR50.72 and 50.73. Connecticut Yankee Atomic Power Company (CYAPCO) and Northeast Nuclear Energy Company (NNECO), on behalf of the Haddam Neck Plant and Millstone Unit Nos. 1, 2, and 3, respectively, commend the Staff for its latest effort to refine its guidance in this area.

CYAPCO and NNECO consider this arena to be one of paramount importance in our day-to-day operation of our nuclear power reactors. We fully appreciate the expanding NRC expectations for us to provide information and reporting on issues that are important to the NRC and to ensure that the Staff is kept fully informed on emerging issues. For the past several years, we have continued our efforts to implement a conservative reporting philosophy in reaching judgements on potentially reportable items. NNECO and CYAPCO appreciate the opportunity to provide our comments on the second draft of the NUREG. In addition to the comments enclosed as Attachment 1, CYAPCO and NNECO have contributed to and endorse the comments that are being submitted by the Boiling Water Reactor Owners' Group and the Nuclear Energy Institute. It is noted that our comments, which are in numerical order with the appropriate page or pages identified, are limited to items of greatest significance to us. Following each comment is the associated page(s) from the NUREG with numerical identifiers corresponding to our comments.

OS3422 REV. 1-94

Very truly yours,

CONNECTICUT YANKEE ATOMIC POWER COMPANY
NORTHEAST NUCLEAR ENERGY COMPANY

FOR: J. F. Opeka
Executive Vice President

BY: [signature]
E. A. DeBarba
Vice President

Attachment

cc: T. T. Martin, Region I Administrator
A. B. Wang, NRC Project Manager, Haddam Neck Plant
J. W. Andersen, NRC Acting Project Manager, Millstone Unit No. 1
G. S. Vissing, NRC Project Manager, Millstone Unit No. 2
V. L. Rooney, NRC Project Manager, Millstone Unit No. 3
W. J. Raymond, Senior Resident Inspector, Haddam Neck Plant
P. D. Swetland, Senior Resident Inspector, Millstone Unit Nos. 1, 2, and 3

U.S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, DC 20555

Docket Nos. 50-213
50-245
50-336
50-423
B14807

Attachment 1

Haddam Neck Plant
Millstone Nuclear Power Station, Unit Nos. 1, 2, and 3
Second Draft NUREG-1022, Revision 1 Comments

April 1994

1) Page 13

NNECO and CYAPCO believe that the example of concern - 17 of 20 main steam safety valves being out of tolerance high - is reportable under § 50.73(a)(2)(i)(B) as an operation or condition prohibited by Technical Specifications.

However, since the relief valves remained capable of performing their intended safety functions, it is not deemed appropriate to report this condition under § 50.72(b)(2)(iii) and § 50.73(a)(2)(v) as a condition that alone could have prevented the fulfillment of a safety function.

Additionally, since the relief valves remained capable of performing their intended safety functions and were thus operable (per Generic Letter 91-18), it is not appropriate to report the subject example per § 50.73(a)(2)(vii) as a common cause failure that caused systems to become inoperable.

Additional Information

Much of the information contained on pages 13 and 14 regarding relief valve testing appears similar to an interpretation of reporting requirements documented in an internal NRC memorandum[1] (provided as Attachment 2 for your convenience). This point is emphasized because it is not optimum to promulgate reporting guidance, much of which is contrary to the established reporting practices of licensees, in the form of an internal NRC memorandum. This is especially true when the NRC resident inspector staff is expecting compliance to an internal memorandum that the licensee does not know exists.

---

(1) E. G. Adensam memorandum to S. J. Collins, "Task Interface Agreement: Interpretation of Reporting Requirements - 93TIA006 (TAC No. M86339)," dated November 2, 1993.

staff, there was no need to report under 10 CFR 50.72 and 50.73 because the NRC was aware of the situation. Some licensee personnel have also expressed a similar understanding for cases in which the NRC staff identified a reportable event or condition to the licensee via inspection or assessment activities. Such means of reporting do not satisfy 10 CFR 50.72 and 50.73. The requirement is to report to the ENS and LER systems events or conditions meeting the criteria stated in the rules.

2.7 Multiple Component Failures

There have been cases in which licensees have not reported multiple, sequentially discovered failures of systems or components occurring during planned testing. This situation was identified as a generic concern on April 13, 1985, in NRC Information Notice (IN) 85-27, "Notifications to the NRC Operations Center and Reporting Events in Licensee Event Reports," regarding the reportability of multiple events in accordance with §§50.72(b)(2)(iii) and 50.73(a)(2)(v) (event or condition that alone could prevent fulfillment of a safety function). [This reporting criterion is discussed in Section 3.3.3 of this report.]

IN 85-27 described multiple failures of a reactor protection system during control rod insertion testing of a reactor at power. One of the control rods stuck. Subsequent testing identified 3 additional rods that would not insert (scram) into the core and 11 control rods that had an initial hesitation before insertion. The licensee considered each failure as a single random failure; thus each was determined not to be reportable. Subsequent assessments indicated that the instrument air system, which was to be oil-free, was contaminated with oil that was causing the scram solenoid valves to fail. While the failure of a single rod to insert may not cause a reasonable doubt that other rods would fail to insert, the failure of more than one rod does cause a reasonable doubt that other rods could be affected, thus affecting the safety function of the rods.

A single component failure in a safety system is reportable if it is determined that the failure mechanism could reasonably be expected to occur in one or more redundant components and thereby prevent fulfillment of the system's safety function. In addition, as indicated in IN 85-27, multiple failures of redundant components of a safety system are sufficient reason to expect that the failure mechanism, even though not known, could prevent the fulfillment of the safety function.

1 Relief Valve Testing

When performing periodic surveillance tests of safety or relief valves it is not uncommon to find more than one valve to be lifting outside of the TS-allowed tolerance band, which is typically plus or minus 1 percent.

If not reportable under §§ 50.72(b)(2)(iii) and 50.73(a)(2)(V) [event or condition that alone could prevent fulfillment of a safety function], this situation would still usually be reportable under §50.73(a)(2)(vii) (common cause failure) because the existence of similar discrepancies in multiple

2) <u>Pages 16 and 17</u>

These additions will clarify the appropriate "Discovery Date" for both self-disclosing reportable events as well as events or conditions which require an evaluation to determine reportability.

2.11 Time Limits for Reporting

10 CFR 50.72

Reporting times in 10 CFR 50.72 are keyed to the occurrence of the event or condition.

- Section 50.72(a)(3) requires ENS notification of the declaration of an Emergency Class "...immediately after notification of the appropriate State or local agencies and not later than one-hour after the time the licensee declares one of the Emergency Classes."

- Section 50.72(b)(1) requires ENS notification for specific types of events and conditions "...as soon as practical and in all cases, within one-hour of the occurrence of any of the following:...."

- Section 50.72(b)(2) requires ENS notification for specific types of events and conditions "...as soon as practical and in all cases, within four hours of the occurrence of any of the following:...."

10 CFR 50.73

THIS IS ESPECIALLY TRUE FOR SELF-DISCLOSING REPORTABLE EVENTS AND CONDITIONS.

10 CFR 50.73 requires submittal of an LER "within 30 days after the discovery" of a reportable event. Many reportable events are discovered when they occur. However, if the event is discovered at some later time, the discovery date is when the reportability clock starts under 10 CFR 50.73.

(2) Discovery date is generally the date when the event was discovered rather than the date when an evaluation of the event is completed. For example, as was discussed in the guidance in NUREG-1022, Supplement 1, Question 14.5, if a technician sees a problem, but a delay occurs before an engineer or supervisor has a chance to review the situation, the discovery date (which starts the 30-day clock) is the date that the technician sees a problem. Thus, for a single event or condition, it is possible to have several applicable dates:

1. The Event Date when the event actually occurred (entered in Item 5 of the LER)

2. The Discovery Date when someone in the plant recognizes that ~~the~~ A REPORTABLE event OR CONDITION has occurred (starts the 30-day clock and should be entered in Item 5 of the LER (event date) if the event date cannot be clearly defined).

3. The Report Date when the LER is submitted (entered in Item 7 of the LER).

The previous guidance in NUREG-1022, Supplement 1, Question 14.5, also discussed a "reportability" date, i.e., the date when someone decides or "discovers" that the event is reportable; however, this date is not used on the LER form or for starting the reportability clock.

If there is a significant length of time (> 30 days) between the event date and either (1) the discovery date or (2) the date when the event was determined to be reportable, the reason for the delay should be discussed in the LER text.

General

In some cases, such as discovery of an existing but previously unrecognized condition, it may be necessary to undertake an evaluation in order to determine if an event or condition is reportable. If so, the guidance provided in Generic Letter 91-18, "Information to Licensees Regarding two NRC Inspection Manual Sections on Resolution of Degraded and Nonconforming Conditions and on Operability," which applies primarily to operability determinations, is appropriate for reportability determinations as well. This guidance indicates that an evaluation should generally proceed on a schedule commensurate with the safety significance of the question. A licensee may continue with plant operation provided there is a reasonable expectation that the equipment in question is operable. Whenever this reasonable expectation no longer exists, or significant doubts begin to arise, the equipment should be considered inoperable and appropriate actions, including reporting, should be taken.

2

Additionally, This is the date that should be documented as the discovery date.

3) Pages 37, 38, and 39

The following enhancements are recommended to emphasize that Systems, Structures, and Components must not only be outside of a conservative design parameter, but must also not be capable of performing its intended safety function (inoperable) to be reportable as outside design basis.

accumulation of voids that could inhibit the ability to adequately remove heat from the reactor core, particularly under natural circulation conditions, would constitute an unanalyzed condition and would be reportable."[6]

"In addition, voiding in instrument lines that results in an erroneous indication causing the operator to misunderstand the true condition of the plant is also an unanalyzed condition and should be reported."[7]

(3) The nuclear power plant being in a condition that is outside the design basis of the plant.

Examples of events or conditions the staff considers reportable include errors in the actual design, such as discovery that an ECCS design does not meet the single failure criterion. They also include hardware problems such as discovery that high energy line break restraints are not installed. In cases such as this, a 10 CFR 50.72 report is sometimes made and then retracted, without submittal of an LER, because further analysis shows that the plant is actually within its design basis. For example, analysis might show that the particular restraints that are missing are not needed for compliance with the design basis.

Another example of an event or condition that the staff considers reportable is discovery that one train of a required two train safety system has been incapable of performing its design function for an extended period of time during operation. For example, in a two-train ECCS system, one train might be found with a design flaw THAT RENDERED THE TRAIN INOPERABLE or with a
3 component that would never have functioned because it was installed incorrectly and a test that would reveal the problem was not performed. This would be considered outside the design basis because, for AN extended period OF time, the system did not have suitable redundancy.[8] Note that this discussion concerns events that actually place the plant outside its design bases. It does not include minor infractions such as (1) cases of technical inoperability, where a component is declared inoperable because a surveillance test is overdue, or (2) cases where the LCO allowed outage time is slightly exceeded. (These conditions may, however, be reportable as conditions prohibited by the Technical Specifications, 10 CFR 50.73(a)(2)(i)(B).)

---

[6] 48 FR 39042, August 29, 1983 and 48 FR 33856, July 26, 1983.

[7] 48 FR 39042, August 29, 1983 and 48 FR 33856, July 26, 1983.

[8] 10 CFR Part 50, Appendix A, Introduction and Criterion 35, and Appendix K, Item I.D.1, indicate that a minimum design criterion is suitable redundancy meeting the single-failure criterion.

(4) The nuclear power plant being in a condition not covered by the plant's operating and emergency procedures.

This criterion points to events where the plant is in a condition outside the coverage of its operating and emergency procedures. A straightforward example of this type of event was the accident at Three Mile Island.

Examples

(1) Maintenance Error

The plant was operating at 100-percent power in steady-state condition. Train "B" essential service water (ESW) system was declared inoperable, depressurized and drained for maintenance. Maintenance technicians were dispatched to loosen train "B" expansion joint in the pipe chase room. The train "A" expansion joint, also located in the pipe chase room, was loosened by mistake as a result of a labelling error and water leaked from the loosened flange joint. The licensee declared train "A" ESW system inoperable and entered TS 3.0.3 because both trains of ESW were inoperable. Repairs were initiated to replace and retorque train "A" expansion joint flange bolts. Train "A" ESW system was declared operable and TS 3.0.3 exited before commencing a plant shutdown.

The licensee made an ENS notification under 10 CFR 50.72(b)(2)(ii)(A) as an unanalyzed condition that significantly compromised plant safety. In a subsequent engineering evaluation the licensee determined that leakage from the loose flange-was insignificant and the flange would remain in place during a design-basis earthquake and, thus, the "A" ESW train was operable and the event was not reportable. However, a voluntary LER was submitted within 30 days.

(2) Unqualified Component

The plant was operating at 100-percent power in steady-state condition. During a review of component classifications, the licensee identified some non-safety-related components which were connected to the drywell (primary containment) safety-related nitrogen supply header. During efforts to upgrade the components to safety-related in accordance with plant procedures, it was determined that certain parts within the non-safety-related components were made of a material that is not suitable for high temperature conditions.

3 It appeared that failure of these parts during post loss of coolant accident (LOCA) conditions could result in the depressurization of the nitrogen supply header and lead to the inability to provide a 100-day supply of nitrogen to safety-related automatic depressurization system (ADS) valves ~~, as described in the updated final safety analysis report (UFSAR)~~. The licensee made an ENS notification because of a condition that placed the plant outside of its design basis. The licensee determined, based on subsequent engineering evaluation, that the maximum

AND THUS RENDERED THE ADS valves INOPERABLE.

3

leakage rate would be less than the capacity of the drywell nitrogen supply header valves, ~~and~~ the 100-day supply of nitrogen was not adversely affected and, thus, the event was not reportable. The ENS notification was retracted.

determined

and

THE SYSTEMS IN QUESTION
REMAINED CAPABLE OF PERFORMING
THEIR INTENDED SAFETY FUNCTION

Docket Nos. 50-213
50-245
50-336
50-423
B14807

Attachment 2

E. G. Adensam Memorandum to S. J. Collins
"Task Interface Agreement: Interpretation of Reporting Requirements"
November 2, 1993

April 1994

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

November 2, 1993

MEMORANDUM FOR: Samuel J. Collins, Director
Division of Reactor Safety
Region IV

FROM: Elinor G. Adensam, Assistant Director
for Regions IV and V
Division of Reactor Projects III/IV/V
Office of Nuclear Reactor Regulation

SUBJECT: TASK INTERFACE AGREEMENT: INTERPRETATION OF REPORTING REQUIREMENTS - 93TIA006 (TAC NO. M86339)

In response to your request dated April 13, 1993, we have reviewed the available guidance associated with the reporting requirements related to multiple failures of safety-related components that are identified during the performance of surveillance procedures. The specific examples cited in your questions regarded the outage surveillances related to primary or secondary safety relief valves and the discovery that the as-found setpoints were outside the allowable technical specification setpoint tolerances. Please note that the Public Document Room (PDR) has been included on the distribution for this response.

Licensees were stated to have presented interpretations of the reporting rules (10 CFR 50.72/50.73) and the related guidance provided in NUREG-1022, which supported the conclusion that the discovery of safety valve setpoint drift was not reportable. Specifically, question 2.3 of NUREG-1022, Supplement 1, had been used to argue that the condition was not reportable, because the condition could be assumed to have occurred at the time of discovery. Another argument presented by licensees was stated to involve analyses or evaluations which determined that the degraded setpoints did not result in the plant operating outside its design basis, and therefore supported a conclusion that the condition was not reportable.

A review of 50.72 and 50.73 identifies several reporting criteria which might be relevant to the discovery of safety valves outside the setpoint tolerances given in the Technical Specifications. These criteria and a discussion of their applicability is provided in Enclosure 1.

The assessment can be summarized as follows:

- The use of question 2.3 to NUREG-1022, Supplement 1, is not appropriate to justify a decision to not report many conditions found during refueling outage surveillances. Other guidance in Supplement 1 is clear that if conditions are discovered during an outage, but are believed to have existed during operation, they are reportable so long as an applicable threshold for reporting is reached.

DF03

1000[1]

002060

- A licensee may determine that a condition such as safety valve setpoint drift, does not constitute operation outside the design basis of the plant, and therefore not report such events in accordance with those criteria in 50.72 and 50.73. However, as discussed below, the condition may be reportable as a result of other criteria.

- 50.73(a)(2)(vii) is deemed the most relevant criterion for the reporting of primary or secondary safety valves found to be outside the acceptable setpoint tolerance. This is due to the fact that this criterion is based on the train or channel level and does not require the loss of a safety function but only the inoperability of multiple channels of a safety system. Some latitude might be given in light of the number of secondary safety valves; but, for most instances of setpoint drift, this criterion would result in the conditions being reportable.

- Note that we currently expect to include guidance along these lines in the forthcoming Revision 1 to NUREG-1022; if so, that specific guidance should be consulted in the future in determining reportability.

Original Signed by

Elinor G. Adensam, Assistant Director
for Regions IV and V
Division of Reactor Projects III/IV/V
Office of Nuclear Reactor Regulation

Enclosure:
Criteria

cc w/enclosure:
B. Hodges, Region I
A. Gibson, Region II
G. Grant, Region III
K. Perkins, Region V

DISTRIBUTION:
Central Files
PDIV-2 Reading
PDR
J. Roe
S. Varga
D. Lovaless, SRI
P. Gwynn, RGN-IV
W. Reckley
G. Holahan
L. Kokajko

*See Previous Concurrence Sheet

Office	PDIV-2/LA	PDIV-2/PM	PDIV-2/PM	AEOD*	PDIV-2/D	EAB*	ADR4/5
Name	EPeyton	LKokajko;nb	WReckley	GHolahan	SBlack	DSkeen	EAdensam
Date	11/2/93	11/2/93	11/2/93	10/22/93	11/2/93	10/27/93	11/2/93
Copy	Yes/No	Yes/No	Yes/No	Yes/No	Yes/No	Yes/No	Yes/No

Document Name: b:\STP\M86339.TIA

ENCLOSURE

ASSESSMENT OF VARIOUS REPORTING REQUIREMENTS FOR APPLICABILITY TO PRIMARY OR SECONDARY SAFETY VALVES FOUND OUTSIDE TECHNICAL SPECIFICATION ACCEPTABLE SETPOINT TOLERANCE BAND

50.72(b)(1)(ii)
50.73(a)(2)(ii)

Any event or condition during operation that results in the condition of the nuclear power plant, including its principal safety barriers, being seriously degraded; or results in the nuclear power plant being:

(A) In an unanalyzed condition that significantly compromises plant safety;
(B) In a condition that is outside the design basis of the plant; or
(C) In a condition not covered by the plant's operating and emergency procedures.

Discussion: The applicability of these criteria is determined by an evaluation of the situation by the licensee. Upon determining that the setpoints were outside the allowable range of the technical specifications, the licensee would be expected to follow the required actions of the technical specifications and assess the plant condition in regards to equipment operability and required corrective actions. Guidance related to the evaluation of degraded and nonconforming conditions is provided by Generic Letter 91-18. As stated in the second draft of NUREG-1022, Revision 1, it is expected that licensees may use engineering judgement and experience in determining whether a condition meets these reporting criteria. The ability of a licensee to justify that a given condition is neither unanalyzed nor outside the design basis is dependent on the as-found condition of the equipment and the degree of analyses performed.

50.72(b)(2)(i) Any event, found while the reactor is shut down, that, had it been found while the reactor was in operation, would have resulted in the nuclear power plant, including its principal safety barriers, being seriously degraded or being in an unanalyzed condition that significantly compromises plant safety.

Discussion: The arguments are very similar to those above and again can support either a reportable or non-reportable conclusion based on the licensee's assessment of the significance of the condition. However, this criterion was intended to capture potential problems which might be discovered only during refueling outage surveillances. Question 7.10 in NUREG-1022, Supplement 1, is considered relevant guidance in regard to the reportability of equipment found to be inoperable during outage surveillances.

Question 2.3 of NUREG-1022, Supplement 1, and the second draft of NUREG-1022, Revision 1, state that failures should be assumed to occur at the time of discovery unless there is firm evidence to believe otherwise. It seems appropriate to classify setpoint drift as a mechanism which would occur some time (usually indeterminable) during the period between calibration and subsequent surveillance unless some factor, such as an extended outage or testing conditions, could be identified as a likely cause. If testing conditions or other causes are identified such that reporting is deemed unnecessary, the licensee would still be expected, under other programs and regulatory requirements, to evaluate the adequacy of the surveillance program to ensure that the activity is ensuring the operability of the safety valves or other components. A voluntary report may still be useful as a means of distributing the information related to the problem and its cause to the industry. Please note that although question 2.3 may be deemed an insufficient reason to determine safety valve drift is not reportable, the licensee may determine that the significance (see above) of the condition does not satisfy the reporting threshold.

50.72(b)(2)(iii)
50.73(a)(2)(v)

Any event or condition that alone could have prevented the fulfillment of a safety function of structures or systems that are needed to:

(A) Shut down the reactor and maintain it in a safe shutdown condition,
(B) Remove residual heat,
(C) Control the release of radioactive material, or
(D) Mitigate the consequences of an accident.

Discussion:

The second draft of NUREG-1022, Revision 1, provides safety valve drift as an example of a common mode problem which may be reportable under this criterion. The example was added to the case described in Information Notice 85-27 which dealt with multiple inoperable control rods. Although certain occurrences of multiple safety valve drift problems should be determined to be reportable under this criterion, it should not be assumed that all cases of one or more safety valves exceeding the technical specification tolerance band need be reportable in accordance with this criterion. As in the previously discussed reporting criteria, the licensee's engineering judgement should determine if the condition could have prevented the fulfillment of a safety function. Candidates for reporting include those cases in which the setpoints of multiple safety valves could have resulted in exceeding the associated system's design pressure. If experience or engineering judgement can reasonably estimate the maximum drift which might occur and determine that the safety function would be maintained, the licensee can determine that the condition is not reportable.

Although discussed in the various drafts and revisions of NUREG-1022, it warrants repeating that the primary motivation behind evaluating plant conditions such as safety valve drift should be to ensure safety and only secondarily to determine reportability. If engineering assessments identify a problem and determine that plant equipment was not and reasonably could not be rendered inoperable by a phenomenon such as setpoint drift, the licensee can then also justify a determination that the condition is not reportable. Voluntary reports are appreciated if the licensee feels the information might be helpful to others. The staff should, as always, be cautious in recommending that a licensee make a "voluntary" report.

50.73(a)(2)(vi) Events covered in paragraph (a)(2)(v) of this section may include one or more procedural errors, equipment failures, and/or discovery of design, analysis, fabrication, construction, and/or procedural inadequacies. However, individual component failures need not be reported pursuant to this paragraph if redundant equipment in the same system was operable and available to perform the required safety function.

Discussion: (See above)

50.73(a)(2)(vii) Any event where a single cause or condition caused at least one independent train or channel to become inoperable in multiple systems or two independent trains or channels to become inoperable in a single system designed to:

(A) Shut down the reactor and maintain it in a safe shutdown condition,
(B) Remove residual heat,
(C) Control the release of radioactive material, or
(D) Mitigate the consequences of an accident.

Discussion: This criterion may be the most relevant to the specific example of safety valves found outside the technical specification tolerance band. As stated in the second draft of NUREG-1022, Revision 1, the reporting threshold for this part of 10 CFR 50.73 is lower than for other parts since it is at the train or channel level rather than the system and function levels. Valves found outside the technical specification setpoint tolerance band can reasonably be considered to have been inoperable during operation unless a licensee determines that testing is not representative of conditions during operation (see item 50.72(b)(2)(i)). This criterion was developed with general consideration given to the normal two train design level of redundancy. Given that most plants can satisfy pressure relief requirements with several main steam safety valves unavailable, a rigid interpretation of this criterion regarding the secondary safety valves (i.e., any case with more than one safety valve outside the tolerance band) may be overly conservative. However, the licensees are considered to have the weakest argument if they determine that this criterion is not applicable, and therefore the condition is not reportable, when finding multiple safety valves outside the acceptable range.

50.73(a)(2)(i.B) Any operation or condition prohibited by the plant's technical specifications.

Discussion: Available guidance regarding operability and technical specification requirements generally have licensees enter the allowed outage time and associated action statements upon discovery of equipment inoperability unless a definite time of inoperability can be established. Technical specifications are considered satisfied provided the allowed outage time and associated action statements are satisfied. Therefore, provided that licensees restore compliance prior to returning to power operation, reporting of safety valve drift in accordance with this criterion would not be necessary. However, it is expected that upon identification of a problem such as safety valve setpoint drift, licensees should take actions to prevent recurrence or pursue a change in the technical specification requirements (such as increasing the acceptable tolerance range of the setpoints). If a licensee determines, through industry experience, information from a vendor, or self assessments, that a component may be inoperable during operation, appropriate actions should be taken in accordance with the technical specifications (reduce power or shutdown). This reporting criterion may be applicable if a licensee fails to satisfy the required action or can determine that a limiting condition of operation had not been satisfied for longer than the allowed outage time following a specific cause for a component becoming inoperable.