DS09
A. Masciantonio

61FR 43092
Aug. 20, 1996

(1)

RECEIVED

Westinghouse
Electric Corporation

Energy Systems

Nuclear Services Division

PO Box 855
Pittsburgh Pennsylvania 15230-0855

1997 JAN -3 PM 1:19

RULES REVIEW & DIR. BR.
USNRC

NSD-NRC-96-4926

December 27, 1996

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

Subject: Comments on the Draft Standard Review Plan (SRP) NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants -- LWR Edition"

Dear Mr. Meyer:

The attached comments are submitted by the Westinghouse Electric Corporation ("Westinghouse") in response to the United States Nuclear Regulatory Commission ("NRC") request for public comments on the proposed draft Standard Review Plan (SRP) NUREG-0800. The NRC request was published in the August 20, 1996 Federal Register, Volume 61, Number 162, page 43092.

Westinghouse appreciates the opportunity to provide these comments. Should you wish to discuss these in greater detail, please contact me at (412) 374-5169.

Very truly yours,

N. J. Liparulo /SAS

N. J. Liparulo, Manager
Regulatory & Engineering Networks

/jas

Comments on SRP DRAFT Rev. 3
Section 15.6.5

1. It is interesting that the NRC has omitted all references to model approval or use of approved models for plant licensing applications, referring instead to consistency with 10 CFR 50, Appendix K or Regulatory Guide 1.157. This is ostensibly consistent with policies promoted and used in recent years by Westinghouse with regard to the use of some LOCA model changes without, or prior to, formal NRC approval. However, it is not entirely consistent with recent experience. The increased flexibility in interpretation, without further clarification, may lead to additional examples in which actions based on a reasonable interpretation of the SRP may lead to Westinghouse exposure based on the NRC's differing expectations. Clarification on this point would be extremely helpful.

2. Clarification would also be helpful in regard to potential reactivity transients arising from unborated water addition through the RCP seals (Section 1, Item 5, page 15.6.5-3). The current Westinghouse models conservatively assume injection of unborated water from all sources, relying (in almost all cases) solely on control rod insertion for reactivity control and plant shutdown. This modeling does not explicitly address the potential for positive reactivity insertion following plant shutdown, and no detailed discussion of SBLOCA core neutronics is typically included in the FSAR. Is the expectation that such evaluation information would be included in the plant FSAR? in addition, this item addressed the "potential for additional core damage". Core damage is not a criterion addressed directly in the LOCA ECCS analyses (not part of 10 CFR 50.46). Does this reflect an expectation to explicitly address core damage as part of the plant FSAR section 15.6.5?

3. Section II (Item 1) and Section IV (Item 1) each identify that "the calculated maximum fuel element cladding temperature does not exceed 1200°C (2200°F). Aside from the strange addition of a metric limit (listed first), 10 CFR 50.46 identifies the limit only as 2200°F, which corresponds to 1204°C (to the nearest °C). Foreign regulators (in countries in which metric or SI units are used in licensing) frequently identify the LOCA PCT limit as 1204°C. While the 4°C is not an overly large difference, it may lead to inconsistency and/or confusion.

4. In several places, review guidance has been added relative to SBLOCA pump trip actions. Most prominently, Section 1, Item 9 states, "...HHFB verifies that the plant operating procedures include actions relative to reactor coolant pump trip following small break loss-of-coolant accidents that are based on plant-specific safety evaluations." The Westinghouse position on RCP pump trip, per generic work documented in WCAP-9584, identifies that the RCP's should be tripped within 10 minutes following the initiation of the SBLOCA. This criterion, in turn, serves as a basis for development of the Westinghouse Emergency Response Guidelines which are

the basis for the plant-specific, event-based Emergency Operating Procedures.

Consequently, this aspect of the SBLOCA analysis is not treated on a plant-specific basis and is not addressed explicitly in the FSAR. The newly added statements in the revised SRP may not be consistent with the current and long-standing Westinghouse position on SBLOCA RCP trip, and may, in essence, introduce a new requirement where none had previously existed in regards to both plant-specific evaluation and documentation expectations.

Comments on Containment and Radiological SRP Sections

Section 6.2.1.3 Mass and Energy Release Analysis for Postulated Loss-of-Coolant Accidents

Page 6.2.1.3-5 Subsection II.B.e PWR Decay Heat Phase, SATAN-V should be replaced with SATAN-VI. The applicable reference for this computer model is:

WCAP-10325-P-A, "Westinghouse LOCA Mass and Energy Release Model for Containment Design March 1979 Version," May 1983.

Note: in subsection II.B.a Subcompartment Analysis, the reference to SATAN-V is applicable.

Section 6.4 Control Room Habitability System

Subsection II.6 Radiation Hazards, the footnote which states that the thyroid doses are consistent with the recommendations of ICRP 26 is incorrect. ICRP 26 lists a 30 rem limit for the dose-equivalent limit to all tissues except the lens. ICRP 26 does not list any specific organ doses, but lists weighting factor for each organ which represents the proportion to the total risk. GDC 19 only lists the limits for the whole body dose of 5 rem.

Section 6.5.1 ESF Atmosphere Cleanup Systems

Subsection II.4 should be GDC <u>43</u> applies ... (editorial)

Section 6.5.2 Containment Spray As A Fission Product Cleanup System

Subsection II.4.a mentions draft NUREG-1465. Reference should be updated to the final version of NUREG-1465. Use of NUREG-1465 for other than CE System 80+ PWR and GE ABWR for fission product may want to be indicated for future considerations.

Section 9.4.2 Spent Fuel Pool Area Ventilation System

Subsection II Technical Rationale 4, under accident conditions per SRP 15.7.4, the criterion is "well within" the 10 CFR Part 100 exposure guidelines, not the 10 CFR Part 20 limits.

Section 15.1.5 Appendix A

Page 15.1.5-3. The requirements of 10CFR Part 100 ... calculating radiation exposure at the site boundary... "and low population zone" should be added here. Loss of a reactor coolant pump does not have any value in calculating the offsite consequences of a steam line break. Therefore this statement should be removed. The section is related to the Radiological Consequences of Main Steam Line Failures Outside Containment of a PWR. Therefore the radiological consequences have no need to consider the containment confinement, and filtering systems. The analysis is typically performed to cover any steam line break whether inside or outside the containment, but assumes that all the faulted steam generator activity is released outside containment.

Page 15.1.5-3 If, as stated previously in the SRP section, that the maximum concentration of radioactive iodine in the coolant is from the Technical Specifications, then the NUREG-1465 specified gap release fractions and fission product release timing are unnecessary. Unless fuel failure is predicted to occur for this event, use of the NUREG-1465 methodology is not needed. The statement of the use of NUREG-1465 should be moved to page 15.1.5-5 item 5.

Page 15.1.5-3, final paragraph under technical rationale. This wording seems to be incomplete. There is no mention of the case with a pre-accident iodine spike.

Throughout: X/Q should be changed to χ/Q.

Section 15.3.3:

Page 15.3.3-7. Is consideration of the containment, confinement and filtering systems needed for this event? There is no release path to the containment since the locked rotor event does not exceed the reactor coolant pressure limit.

Section 15.4.8 Appendix A

Page 15.4.8-3. Consistency on the limits of the rod ejection accident doses are needed. The limits as defined in the Acceptance Criteria is "well within" the 10CFR 100.11 exposure guideline values. Subsection II.2 states within the 10 CFR 100.11 guidelines for leak rate. Technical Rationale part 1 and 2 also imply the use of the 10 CFR 100.11 guidelines. These limits must be consistent or defined adequately.

Page 15.4.8-3, under Technical Rationale part 2. The rod ejection accident is not the limiting event for the primary to secondary leak rate. This is either the locked rotor or the steamline break events. The rod ejection accident is dominated by the portion of activity released via containment leakage. Therefore it is unclear why a more stringent primary to secondary leakage rate could be necessary based on the results of this transient. Reword or delete this section.

Section 15.6.2

Page 15.6.2-3, II.2: 10% of the 10CFR100.11 guidelines is defined elsewhere in the SRPs as a "small fraction". Why change the definition to "substantially below" for this one section. Remain consistent throughout the SRP sections.

Page 15.6.2-4, item 2. The first sentence needs the word "from" in (a) a fission product release from the plant......

Page 15.6.2-8. First sentence says the offsite consequences are within the guidelines of 10 CFR Part 100. The criteria is 10% of 10CFR100.11. Provide consistency.

Section 15.6.3

Section 15.6.3 is assumed to apply to future plant licensing applications not for plants which are currently operating. The currently operating Westinghouse plants which were licensed prior to 1981 typically do not meet all the requirements of this SRP section. Typically the licensing basis thermal and hydraulic calculations for these plants do not consider a limiting single failure or provide a specific analysis to determine margin to steam generator overfill. These calculation do however consider loss of offsite power since this provides the limiting offsite radiological consequences. Future plant applications and operating plants licensed after 1981 do meet the interpretation of this SRP section.

Page 15.6.3-2 states that the evaluation of the radiological consequences should consider a limiting single failure and with loss of offsite power. The limiting single failure and loss of offsite power are typically incorporated into the thermal and hydraulic calculation of the primary to secondary break flow and steam releases to the atmosphere resulting from a steam generator tube rupture. The offsite dose calculation uses these results to determine the consequences of the event. Clarification of this point is needed in this section.

Page 15.6.3-2, Review Interfaces no. 1 should also explain that the thermal and hydraulic analysis is used to determine when the break flow is terminated and if margin to steam generator overfill exists.

Page 15.6.3-4 Meeting the requirements ...... There are two limits for the SGTR event with pre-accident and concurrent accident spike. These should be defined here.

Page 15.6.3-6, item 10. It should be noted here that this is performed in the thermal and hydraulic calculation.

Page 15.6.3-7, item 14. It is assumed that this SRP section applies to new plant applications.

Page 15.6.3-8, last paragraph, the limits for both iodine spike cases should be listed.

Generic Comment on section: The section consistently does not differentiate between the calculation of the primary to secondary break flow and steam releases and the calculation of the offsite doses. There is a need to define which calculation is being referred to in each section.

Section 15.6.5 Appendix A

This section does not refer to the use of NUREG-1465 source term methodology as the other SRP sections which have fuel damage. This should be incorporated into the LOCA section also for future applications.

Page 15.6.5-8, end of second paragraph. The statement "are well within these guidelines" indicates the limits are 25% of the guidelines. This is not correct. Statement should be "are within the guidelines".

Section 15.7.4

Page 15.7.4-4. The format of 10 CFR Part 100, section 100.11 is not consistent with other SRP sections as 10 CFR 100.11. Consistency is needed throughout this Section.

Page 15.7.4-9, Reference 7 needs completion date.

Non-LOCA Transient Analysis
Comments on draft SRP

General Comments

1. The following two sentences have been added to most of the SRP sections, under subsection V., Implementation:

> "This SRP section will be used by the staff when performing safety evaluations of license applications submitted by applicants pursuant to 10 CFR 50 or 10 CFR 52."
>
> "The provisions of this SRP section apply to reviews of applications docketed six months or more after the date of issuance of this SRP section."

Westinghouse interprets these statements to mean that any revised SRP requirements will be applicable to new license applications only, and not to license amendment requests for currently licensed plants. This interpretation is consistent with past precedent. For licensing amendments, older plants have generally not been required to analyze accidents or apply new assumptions that were not originally required in their licensing basis. This could be clarified by the use of the term "new license applications" instead of just "license applications."

2. The accident analysis SRP sections have been updated to include a "Technical Rationale" section, which is intended to provide the basis for the applicability of the identified acceptance criteria (GDCs) for each event. In many instances these statements should be improved to be more accurate and meaningful. As an example, in SRP section 15.2.6, the technical rationale given for GDC 15 includes the following statements:

> "Compliance with GDC 15 requires that the reactor coolant system and associated auxiliary, control, and protection systems be designed with sufficient margin to ensure that the design conditions of the reactor coolant system pressure boundary are not exceeded during any condition of normal operation, including anticipated operational occurrences."
>
> "GDC 15 is applicable to this section because the reviewer evaluates the consequences of the loss of nonemergency ac power to the station auxiliaries. This is an anticipated operational occurrence, and the reactor coolant pressure needs to be analyzed to ensure that the pressure acceptance critierion is satisfied."
>
> "Meeting the requirements of GDC 15 provides assurance that the design conditions of the reactor coolant pressure boundary are not exceeded for the loss of nonemergency ac power to the station auxiliaries."

The first paragraph above accurately summarizes the requirements of the GDC. The first sentence of the second paragraph is not accurate or meaningful. It says that the GDC is

applicable because the reviewer is evaluating the accident. This point is irrelevant to GDC applicability. The GDC is applicable because the event under consideration is an anticipated operational occurrence for which the potential exists to exceed the design conditions of the RCS pressure boundary. Finally, the third paragraph is a circular and redundant restatement of the GDC requirements, which adds no value to the text. It could be summarized as saying, "Meeting the requirements of GDC 15 provides assurance that [the requirements of GDC 15 are met.]" This generally applies to the final paragraph of all of the Technical Rationale subsection items.
It is recommended that all of the Technical Rationale sections be revised to provide a meaningful and accurate basis for the GDC applicability.

3. As much as possible, the format and content of the individual accident analysis SRP sections should be standardized and consistent. For instance:

a. There are instances where different numbering schemes are used. For example, under "Technical Rationale," the numbering of the paragraphs uses (a), (b), ... in some sections (e.g., 15.1.1, 15.2.6), and 1., 2., ... in other sections (e.g., 15.1.5, 15.2.7).

b. Subsections are not numbered consistently in different SRP sections. For example, in SRP section 15.2.7, under II. Acceptance Criteria, after the GDC listing, the text is divided with subsections numbered 1., 2., ... and sub-subsections a., b., ... Subsection 1. lists basic objectives of the review, subsection 2 lists specific criteria, etc. But in other SRP sections, e.g., 15.2.6, this numbering scheme is not used. No basic objective for the review section is present, and there is no well-defined subsection numbering. The text goes right from sections numbered 1., 2., ... to others numbered a., b., ...

c. Several accident analysis SRP sections provide lists of parameters for which time-related variations should be reviewed. Other SRP sections do not provide such a list. For those that do provide a list, the format of the list is not consistent from section to section. In addition, these parameter lists generally contain some parameters which are not meaningful for the transient under consideration and are not typically presented in SAR sections prepared by Westinghouse. At the same time, other parameters which are important are sometimes not included.

d. Most of the accident analysis SRP sections provide a reference to various acceptable analytical models for the various vendors plants. However, in identifying these acceptable models, some SRP sections refer to the vendor's standard reference plant SAR (e.g., 15.1.1-15.1.4), others refer to specific analyis computer code topicals (e.g., 15.2.1), and still others refer to the NRC's SER for the standard reference SAR (e.g., 15.2.6).

e. Different SRP sections designate footnotes variously, using either letters, numbers, or symbols.

Westinghouse Transient Analysis Specific SRP Section Comments

SRP Section No.: **15.0** Subject: **Introduction**		
Subsection #	SRP Page #	Comment
N/A	15.0-3	In sixth paragraph, change "doppler" to "Doppler," consistent with usage throughout the SRP.

SRP Section No.: **15.1.1 - 15.1.4** Subject: **Decrease in Feedwater Temperature, Increase in Feedwater Flow, Increase in Steam Flow, and Inadvertent Opening of a Steam Generator Relief or Safety Valve**		
Subsection #	SRP Page #	Comment
I. Areas of Review	15.1.1-2	Under Review Interfaces, section A., regarding HICB, it is stated "... as part of its primary review responsibility for SRP Sections 7.2 through 7.5." In the same discussion about HICB for other SRP sections these SRP Section numbers are sometimes listed differently, such as "7.1 through 7.7" or "7.2 through 7.7." Are these meant to be different, or should all SRP sections be consistent?
II. Acceptance Criteria	15.1.1-3	It is not clear why GDC 17 has not been added to this section, as it has been to others. The integrated impacts statements for adding GDC 17 indicate that "GDC 17 ... should be added to those sections where turbine trip results from the transient." This would apply to some of these transients.
II. Acceptance Criteria	15.1.1-5	In Item c., change "doppler" to "Doppler," consistent with usage throughout the SRP.
IV. Evaluation Findings	15.1.1-10	In the introductory paragraph beginning "The staff concludes...", reference to TMI Action Plan items II.E.5.1 and II.E.5.2 has been deleted, however, paragraph number 5 at the bottom of the page still refers to these item numbers.
V. Implementation	15.1.1-11	The last word in this section should be changed from "NUREGS" to "NUREGs."

SRP Section No.: **15.1.5**
Subject: **Steam System Piping Failures Inside and Outside of Containment**

Subsection #	SRP Page #	Comment
I. Areas of Review	15.1.5-4	In the first new sentence on the page, correct spelling from "axillary" to "auxiliary."
II. Acceptance Criteria	15.1.5-5	Two occurrences of the term "Task Action Plan" ; other SRP sections (e.g., 15.1.1) use the term "TMI Action Plan." Both are in reference to NUREG-0737 items. Consistent terminology should be used throughout the SRP if these are the same.
IV. Evaluation Findings	15.1.5-13	In paragraph (f), reference is made to X/Q. Some other SRP sections have been updated to use the greek character in the expression, χ/Q. All SRP sections should be updated consistently. In addition, the units on X/Q in this paragraph should be corrected from "sec/m3$" to "sec/m^3."

SRP Section No.: **15.1.5, Appendix A**
Subject: **Radiological Consequences of Main Steam Line Outside Containment of a PWR**

Subsection #	SRP Page #	Comment
all	all	Page numbering for this section is redundant to the main section 15.1.5. Page numbers should be changed to 15.1.5A.x to avoid confusion.

SRP Section No.: **15.2.1-15.2.5**
Subject: **Loss of External Load; Turbine Trip; Loss of Condenser Vacuum; Closure of Main Steam Isolation Valve (BWR); and Steam Pressure Regulator Failure (Closed)**

Subsection #	SRP Page #	Comment
I. Areas of Review	15.2.1-2	Item 4, Main Steam Isolation Valve Closure, and the title of this SRP section imply that this event is only applicable to a BWR. However, MSIV closure can occur for a PWR as well. This event is bounded by Turbine Trip for a Westinghouse PWR.
II. Acceptance Criteria	15.2.1-4	It is not clear why GDC 20 does not apply to these events, which are AOOs. Integrated impact 1351 added GDC 20 to section 15.1.1-15.1.4 on the basis that it is applicable to anticipated operational occurrences.
	15.2.1-7	In the first line on the page, change "SRP Section 15.1.5" to "SRP Section 15.2.1–15.2.5"

	15.2.1-7	In the second paragraph, change "result initiating" to "result of initiating "
IV. Evaluation Findings	15.2.1-10	In the text, reference to footnote number 1 was deleted but the footnote itself is not. Delete footnote number 1 and renumber footnote number 2.
	15.2.1-10	The new paragraph of text beginning on this page, starting with "The transient ..." has a right margin that is not consistent with the paragraphs immediately above and below.
VI. References	15.2.1-12	Reference 5 should be updated to the following: Burnett, T.W.T., et al., "LOFTRAN Code Description," WCAP-7907-P-A (proprietary), and WCAP-7907-A (nonpropietary), Westinghouse Electric Corporation, April 1984.
SRP Section No.: **15.2.6** Subject: **Loss of Nonemergency AC Power to the Station Auxiliaries**		
Subsection #	SRP Page #	Comment
I. Areas of Review	15.2.6-1	In first paragraph, change the reference to the loss of load SRP section number from "15.2.2" to "15.2.1-15.2.5."
II. Acceptance Criteria	15.2.6-6	In the "Technical Rationale" for GDC 26, item (c), second paragraph, it is stated that GDC 26 is applicable because "the transient analyzed in this section will involve the movement of control rods in response to the transient..." Since this event is characterized by an early reactor trip, rod control is not an important consideration. It is recommended that the text be revised; the corresponding paragraph contained in the loss of normal feedwater SRP section 15.2.7, page 15.2.7-8 is more appropriate for use here.
SRP Section No.: **15.2.7** Subject: **Loss of Normal Feedwater Flow**		
Subsection #	SRP Page #	Comment
		None.

SRP Section No.: **15.2.8**
Subject: **Feedwater System Pipe Breaks Inside and Outside Containment (PWR)**

Subsection #	SRP Page #	Comment
II. Acceptance Criteria	15.2.8-8	In paragraph h., there appears to be a typographical problem with the word "for" before "operator action."
	15.2.8-9 through 15.2.8-11	The second paragraph under item 2 (GDC 17) technical rationale, states that feedwater system pipe breaks can be classed as AOOs or accidents, depending on severity. Westinghouse does not consider a feedwater system pipe break to be an "anticipated operational occurrence," but rather a "postulated accident." Furthermore, shouldn't the classification of an event be based on the probability of occurrence rather than the severity of the event? Perhaps "severity" here refers to the break itself and not the event consequences. This should be clarified.

SRP Section No.: **15.3.1 - 15.3.2**
Subject: **Loss of Forced Reactor Coolant Flow Including Trip of Pump Motor and Flow Controller Malfunctions**

Subsection #	SRP Page #	Comment
II. Acceptance Criteria	15.3.1-4	In the introductory text beginning "The specific criteria ..." GDC 17 should be added to the list.
	15.3.1-4	In item a., the reference number should be changed from "Ref. 5" to "Ref. 6," since the reference number for the ASME code has been changed.
	15.3.1-4	In the last paragraph on this page, second sentence, the text has been correctly changed from the previous "References 6 through 9" to "References 7 through 10," but this has not been properly identified as a change via redlining and shading. Similarly, in footnote a, "References 10 and 11" has been changed to "References 11 and 12" without identifying the change. Also, footnotes are designated with numbers instead of letters in other SRP sections.
VI. References	15.3.1-10	This is rather trivial, but the quote marks contained in the new Reference 3 are of a different form than all the other quote marks in the text. Should be consistent.

SRP Section No.: **15.3.3 - 15.3.4**
Subject: **Reactor Coolant Pump Rotor Seizure and Reactor Coolant Pump Shaft Break**

Subsection #	SRP Page #	Comment
I. Areas of Review	15.3.3-3	In paragraph 4, the "SPLB" branch name should be spelled out, consistent with the format used for the other branch names. This is the first use of this acronym in this SRP section.
II. Acceptance Criteria	15.3.3-3	In newly added paragraph A, change "General Design Criterion (GDC) 17" to "General Design Criterion 17 (GDC 17)," for consistency with the other paragraphs.
	15.3.3-4	In the introductory text beginning "The specific criteria ..." GDC 17 should be added to the list.
	15.3.3-4	In newly numbered paragraph 7, change "(see Refs. 5 and 6)" to "(see Refs. 6 and 7)," to reflect updated reference numbers.
	15.3.3-5	In the large paragraph following newly numbered paragraph 9, second sentence, the text has been correctly changed from the previous "References 7 through 13" to "References 8 through 12," but this has not been properly identified as a change via redlining and shading.
	15.3.3-6	In about the middle of the first paragraph on this page, change "abnormal operating occurrences" to "anticipated operational occurrences," consistent with the GDC terminology.
III. Review Procedures	15.3.3-8	In item number 6, change the parenthetical "(see II.3.b)" to "(see II.ii)." Note that the subsection II numbering is not consistent with other SRP sections and is not sufficiently structured to permit unambiguous specific referencing from other subsections of the document. See General Comment number 3.b above.
III. Review Procedures, and IV. Evaluation Findings	15.3.3-7, 15.3.3-10	The underlining on these section headings incorrectly extends to the blanks between the section number and the title.
V. Implementation	15.3.3-11	In the last paragraph of this section, change "Subsection" to "subsection" (two occurrences) consistent with the form used elsewhere.

VI. References	15.3.3-11	This is rather trivial, but the quote marks contained in the new Reference 1 are of a different form than all the other quote marks in the text. Should be consistent.
	15.3.3-12	Newly numbered Reference 8 should be revised to the following, which is an approved Westinghouse code for analyzing this event. The methodology of WCAP-7973 is fully incorporated into LOFTRAN. Burnett, T.W.T., et al., "LOFTRAN Code Description," WCAP-7907-P-A (proprietary), and WCAP-7907-A (nonpropietary), Westinghouse Electric Corporation, April 1984.
Attachment A - Table of Proposed Changes	15.3.3-15	In the desription of item 13, the acronyms HICB and ICSB are reversed.
	15.3.3-15	In the description of items 16 and 17, the acronym TERB should be PERB.
	15.3.3-17	In the description of item 36, change "References 14 through 18" to "References 15 through 19."
	15.3.3-17	In the description of item 37, change "Reference 19" to "Reference 20."
	15.3.3-17	In the description of item 48, the acronyms HICB and ICSB are reversed.
	15.3.3-18	In the description of item 53, the acronym TERB should be PERB.

SRP Section No.: **15.4.1**
Subject: **Uncontrolled Control Rod Assembly Withdrawal from a Subcritical or Low Power Startup Condition**

Subsection #	SRP Page #	Comment
II. Acceptance Criteria	15.4.1-2	In paragraph number 2 starting with "The requirements ..." GDC 17 should be added to the series of numbers.
IV. Evaluation Findings	15.4.1-5	In the sample staff conclusion paragraphs nothing is explicitly stated for the newly added GDC 17 as it is for the other GDCs.

SRP Section No.: **15.4.2**
Subject: **Uncontrolled Control Rod Assembly Withdrawal at Power**

Subsection #	SRP Page #	Comment
III. Review Procedures, and IV. Evaluation Findings	15.4.2-4 through 15.4.2-7	In these sections, it appears there has been a global replacement of the word "transient" with "AOO." While this may be intended and appropriate in a few instances, it does not appear to be appropriate or correct for several of these. For example, see paragraph 3 on page 15.4.2-5, where "nuclear transient" becomes "nuclear AOO." Review all these instances to ensure the change is intended.

SRP Section No.: **15.4.3**
Subject: **Control Rod Misoperation (System Malfunction or Operator Error)**

Subsection #	SRP Page #	Comment
II. Acceptance Criteria	15.4.3-4	In the second paragraph of item 2, the term "reactivity control system" is given the acronym RCS, which is then used only one time after that in the second paragraph of item 3. Since this acronym is commonly used for "reactor coolant system" its use here as something else could cause confusion. Recommend deleting this acronym and just spelling it out one more time.
III. Review Procedures	15.4.3-5	In paragraph 2, "transient methods" is inappropriately changed to "AOO methods" - another case of global replacement (see SRP section 15.4.1 comment above.)

SRP Section No.: **15.4.4-15.4.5**
Subject: **Startup of an Inactive Loop or ...**

Subsection #	SRP Page #	Comment
I. Areas of Review	15.4.4-1	The term "moderate frequency" in the first sentence should reference ANS-51.1/N18.2-1973 which classifies the events into the categories of Condition I (Normal Operation), II (Incidents of Moderate Frequency), III (Infrequent Incidents) and IV (Limiting Faults) and which specifies design requirements. The term "anticipated operational occurrences (AOOs)" as defined in 10CFR50 Appendix A does not distinguish between Condition I, II and II events although the majority of SARs do when discussing the different events. A more clear definition to be used throughout the SRP section (and other SRP sections) might be "AOOs of moderate frequency".

III. Review Procedures	15.4.4-8	In item 6, change "(see III.3.b)" to "(see II.b).". Note that the subsection II numbering is not consistent with other SRP sections and is not sufficiently structured to permit unambiguous specific referencing from other subsections of the document. See General Comment number 3.b above.
SRP Section No.: **15.4.6** Subject: **Chemical and Volume Control System Malfunction that Results in a Decrease in Boron Concentration in the Reactor Coolant (PWR)**		
Subsection #	SRP Page #	Comment
		None.
SRP Section No.: **15.4.7** Subject: **Inadvertent Loading and Operation of a Fuel Assembly in an Improper Position**		
Subsection #	SRP Page #	Comment
		None.
SRP Section No.: **15.4.8** Subject: **Spectrum of Rod Ejection Accidents (PWR)**		
Subsection #	SRP Page #	Comment
III. Review Procedures	15.4.8-5	In paragraph 3.b, last line, change "W-3 (Ref. 4) or BAW-2 (Ref. 5)" to "W-3 (Ref. 5) or BAW-2 (Ref. 6)" consistent with updated reference numbering.
IV. Evaluation Findings	15.4.8-6	In the middle of the first paragraph, use the letter "O" instead of the number "0" in "UO_2."
SRP Section No.: **15.4.8, Appendix A** Subject: **Radiological Consequences of a Control Rod Ejection Accident (PWR)**		
Subsection #	SRP Page #	Comment
all	all	Page numbering for this section is redundant to the main section 15.4.8. Page numbers should be changed to 15.4.8A.x to avoid confusion.
IV. Evaluation Findings	15.4.8-5	The first paragraph in this section has an improper right margin.

SRP Section No.: **15.5.1 - 15.5.2**
Subject: **Inadvertent Operation of ECCS and Chemical and Volume Control System Malfunction that Increases Reactor Coolant Inventory**

Subsection #	SRP Page #	Comment
I. Areas of Review	15.5.1-1	In footnote 1, "Reference 8" should be "Reference 16" and "Reference 9" should be "Reference 17," to match the previous SRP. Note that major revisions were made to the references for this SRP section. Care should be taken to ensure that these are all referenced properly.
III. Review Procedure	15.5.1-7	In item 6, change "(see II.3.b)" to "(see II.b)." (?) However, note that the subsection II numbering is not consistent with other SRP sections and is not sufficiently structured to permit unambiguous specific referencing from other subsections of the document. In this case there are two separate items "b" in subsection II. See General Comment number 3.b above.

SRP Section No.: **15.6.1**
Subject: **Inadvertent Opening of a PWR Pressurizer Pressure Relief Valve or a BWR Pressure Relief Valve**

Subsection #	SRP Page #	Comment
		None.