Palo Verde Nuclear
Generating Station

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April 1, 2001

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

Dear Sirs:

**Subject: Palo Verde Nuclear Generating Station (PVNGS)
Units 1, 2 and 3
Docket Nos. STN 50-528/529/530
Request for Amendment to Various Administrative Controls for Section 5.0 of Technical Specifications**

Pursuant to 10 CFR 50.90, Arizona Public Service Company (APS) requests an amendment to various administrative controls of Technical Specification (TS) Section 5.0 for Palo Verde Nuclear Generating Station (PVNGS) Units 1, 2, and 3. This proposed amendment includes the following Technical Specification changes:

1. TS 5.5.13, Diesel Fuel Oil Testing Program – Clarify that the limits for water and sediment content of new diesel fuel oil are not in ASTM D1796.
2. TS 5.5.14, Technical Specifications (TS) Bases Control Program – Revise guidance for making changes to TS Bases without prior NRC approval based on changes to 10 CFR 50.59.
3. TS 5.5.15, Safety Function Determination Program (SFDP) – Add clarification to the requirements for the SFDP.
4. TS 5.6.5, Core Operating Limits Report (COLR) – Add CENTS computer code to the list of analytical methods used to determine the core operating limits.
5. TS 5.6.5, Core Operating Limits Report (COLR) – Revise the list of referenced approved topical reports to be sited using the report number and title.

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Provided in Enclosure 1 to this letter are the following sections which support the proposed Technical Specification amendment:

A. Description of the Proposed Technical Specification Amendment
B. Purpose of the Technical Specification
C. Need for the Technical Specification Amendment
D. Safety Analysis for the Proposed Technical Specification Amendment
E. No Significant Hazards Consideration Determination
F. Environmental Consideration
G. Marked-up Technical Specification Pages
H. Retyped Technical Specification Pages

Enclosure 2 contains the proposed changes to the bases for Limiting Condition for Operation 3.0.6. These changes are provided for information only to support the review of the associated TS change to the Safety Function Determination Program.

In accordance with the PVNGS Quality Assurance Program, the Plant Review Board and Offsite Safety Review Committee have reviewed and concurred with this proposed amendment. By copy of this letter this request is being forwarded to the Arizona Radiation Regulatory Agency (ARRA) pursuant to 10 CFR 50.91(b)(1).

APS requests that the enclosed Technical Specification amendment request be reviewed and approved by September 1, 2001 since that is when PVNGS plans to start to use the CENTS computer code for core design calculations. The implementation of the core design is planned for Unit 2, cycle 11 in April 2002. It is requested that this proposed amendment become effective 60 days following the issuance of this amendment.

No commitments are being made to the NRC by this letter.

Should you have any questions, please contact Scott A. Bauer at (623) 393-5978.

Sincerely,

David Mauldin

CDM/SAB/JAP/kg

Enclosures

cc: E. W. Merschoff
J. N. Donohew
J. H. Moorman
A. V. Godwin (ARRA)

STATE OF ARIZONA)
) ss.
COUNTY OF MARICOPA)

I, David Mauldin, represent that I am Vice President Nuclear Engineering and Support, Arizona Public Service Company (APS), that the foregoing document has been signed by me on behalf of APS with full authority to do so, and that to the best of my knowledge and belief, the statements made therein are true and correct.

David Mauldin

David Mauldin

Sworn To Before Me This 2nd Day Of April, 2001.

Nora Meador

Notary Public

"OFFICIAL SEAL"
Nora E. Meador
Notary Public-Arizona
Maricopa County
My Commission Expires 4/6/2003

Notary Commission Stamp

ENCLOSURE 1

Proposed Amendment to Units 1, 2 and 3 Technical Specifications, Various Administrative Controls to Section 5.0

A. DESCRIPTION OF THE PROPOSED TECHNICAL SPECIFICATION AMENDMENT

The proposed amendment will change the following specifications in section 5.0 of the Technical Specification (TS) for Palo Verde Nuclear Generating Station (PVNGS) Units 1, 2, and 3:

TS 5.5.13.a.3, "Diesel Fuel Oil Testing Program" –

This change will correct an inconsistency between this technical specification, technical specification bases and the Updated Final Safety Analysis Report (UFSAR) concerning the water and sediment content testing standard for new diesel fuel oil. TS 5.5.13.a.3 will be changed to state; "Water and sediment within limits when tested in accordance with ASTM D1796."

TS 5.5.14.b, "Technical Specifications (TS) Bases Control Program" –

The criteria for making changes to TS Bases without prior NRC approval has been revised based on changes to 10 CFR 50.59 rule. This will entail two changes. The first change will be to the initial sentence for TS 5.5.14.b. The word "involve" will be replaced with "require". The next change will alter the second item listed for TS 5.5.14.b to state, "A change to the updated FSAR or Bases that requires NRC approval pursuant to 10 CFR 50.59." These changes are consistent with NRC approved Technical Specifications Task Force (TSTF) traveler number 364-revision 0. Additionally, the two criteria of TS 5.5.14.b will be numerically formatted.

TS 5.5.15, "Safety Function Determination Program (SFDP)" –

Clarification is being added to the requirements for implementing the SFDP.

The second paragraph of TS 5.5.15 will be changed to read:

"A loss of safety function exists when, assuming no concurrent single failure, no concurrent loss of offsite power, or no concurrent loss of onsite diesel generator(s), a safety function assumed in the accident analysis cannot be performed. For the purpose of this program, a loss of safety function may exist when a support system is inoperable, and:"

An additional paragraph will be added to the end of TS 5.5.15 stating:

"When a loss of safety function is caused by the inoperability of a single Technical Specification support system, the appropriate Conditions and Required Actions to enter are those of the support system."

Additionally, clarification will be added to Limiting Condition for Operation (LCO) 3.0.6 Bases of the "appropriate LCO for loss of safety function." The Bases will also clarify the requirements for the SFDP that consideration does not have to be made for a loss of power in determining loss of function. This change is consistent with NRC approved TSTF traveler number 273-revision 2, as amended by Westinghouse Owners Group (WOG) editorial change WOG-ED-23.

In addition, an editorial change to remove the "s" from the word "Functions" in the title for TS 5.5.15 will occur. This change reflects the plant specific name for this program.

TS 5.6.5.b, "Core Operating Limits Report (COLR)" -

The CENTS computer code will be added to the list of analytical methods used to determine the core operating limits. TS 5.6.5.b.13 will add the option to use the CENTS computer code in licensing analyses in place of the currently licensed CESEC code. The CENTS computer code has been generically approved by the NRC for the calculation of transient behavior in Pressurized Water Reactors (PWRs) designed by Combustion Engineering (CE) with some limitations.

One of the limitations associated with the NRC safety evaluation for the generically approved CENTS code (reference 1) involves the use of CENTS for CEA ejection analyses. In Section C, APS provides clarification of this limitation based on the topical report submittal and our understanding of the limitation.

PVNGS intends to qualify CENTS for use in future Palo Verde licensing analyses by following the guidelines prescribed in Generic Letter (GL) 83-11, Supplement 1. TS 5.6.5.b. will have the CENTS code listed as the following:

"Technical Manual for the CENTS Code," CE-NPD 282-P-A, Volumes 1-3, [Methodology for Specifications 3.1.2, Shutdown Margin-Reactor Trip Breakers Closed; 3.1.4, Moderator Temperature Coefficient; 3.1.5, CEA Alignment; 3.1.7, Regulating CEA Insertion Limits; 3.1.8, Part Length CEA Insertion Limits and 3.2.3, Azimuthal Power Tilt - T_q]."

TS 5.6.5.b, "Core Operating Limits Report (COLR)" -

The list of referenced reports for analytical methods used is being revised to identify only the report number and title. Additionally, the following note will be added to 5.6.5.b to ensure that the full identification of the referenced approved reports are contained in the COLR.

"The COLR will contain the complete identification for each of the Technical Specification referenced topical reports used to prepare the COLR (i.e., report number, title, revision, date, and any supplements)."

Additionally, TS 5.6.5.b.6 and 5.6.5.b.7 both list the same topical report (Calculative Methods for the CE Small Break LOCA Evaluation Model, CENPD-137). TS 5.6.5.b.7 is the supplement to the topical report listed in 5.6.5.b.6. TS 5.6.5.b.7 will be deleted and the "Calculative Methods for the CE Small Break LOCA Evaluation Model, CENPD-137" (along with its supplement) will be listed in full text within the COLR. For informational purposes, Enclosure 4 contains the full text version of those topical reports that are being modified in TS 5.6.5 and will subsequently be listed in the PVNGS COLR.

This change is consistent with the NRC accepted TSTF traveler number 363 – revision 0 and NRC's letter, "Acceptance for Siemens References to Approved Topical Reports in Technical Specifications", dated December 15, 1999 (reference 1).

B. PURPOSE OF THE TECHNICAL SPECIFICATION

TS 5.5.13, Diesel Fuel Oil Testing Program, establishes the program used to implement required testing of both new and stored diesel fuel oil. This program includes sampling and testing requirements, and acceptance criteria, all in accordance with applicable American Society for Testing and Materials (ASTM) Standards as referenced in the Updated Final Safety Analysis Report (UFSAR).

TS 5.5.14, TS Bases Control Program, provides the requirement for maintaining a program for processing changes to the Bases of the Technical Specifications.

TS 5.5.15, SFDP, ensures loss of safety functions are detected and appropriate actions are taken. This program implements the requirements of Limiting Condition for Operation (LCO) 3.0.6.

TS 5.6.5.b, COLR, lists those analytical methods used to determine the core operating limits. These analytical methods are reviewed and approved by the NRC.

C. NEED FOR THE TECHNICAL SPECIFICATION AMENDMENT

TS 5.5.13, Diesel Fuel Oil Testing Program. TS 5.5.13.a.3 currently states, "Water and sediment are within the limits of ASTM D1796," for the acceptability of new fuel oil. This is an incorrect reference for the limits of water and sediment content for new diesel fuel oil. The water and sediment limits for new fuel oil are contained within the Technical Specification Bases. ASTM D1796 contains testing methods used for analysis of new fuel oil for water and sediment. The acceptability of new diesel fuel oil for use, concerning its water and sediment content, in TS 5.5.13.a.3 will be changed to state, "Water and sediment within limits when tested in accordance with ASTM D1796."

TS 5.5.14, TS Bases Control Program, requires a program for processing changes to the Bases of the Technical Specifications. TS 5.5.14.b states,

> "Licensees may make changes to Bases without prior NRC approval provided the changes do not involve either of the following:
>
> > A change in the TS incorporated in the license; or
> >
> > A change to the updated FSAR or Bases that involves an unreviewed safety question as defined in 10 CFR 50.59."

In the initial sentence to TS 5.5.14.b, the word "involve" will be replaced with "require". Additionally, the second allowance for changing TS Bases as described in TS 5.5.14.b will be revised to state, "A change to the updated FSAR or Bases that requires NRC approval pursuant to 10 CFR 50.59." Both of these changes are based on the changes to 10 CFR 50.59 published in the Federal Register (Volume 64, Number 191) dated October 4, 1999. This change is consistent with NRC approved TSTF traveler number 364-revision 0.

This change will also numerically format the two options listed in TS 5.5.14.b. This is consistent with other listings contained in Section 5.0 of the TS.

TS 5.5.15, SFDP. The NRC has approved TSTF 273-revision 2, as amended by editorial change WOG-ED-23. This TSTF makes changes to TS 5.5.15 and TS Bases for LCO 3.0.6. The first change to TS 5.5.15 will add words to the second paragraph. The first sentence of this paragraph will now state: "A loss of safety function exists when, assuming no concurrent single failure, **no concurrent loss of offsite power, or no concurrent loss of onsite diesel generator(s),** a safety function assumed in the accident analysis cannot be performed." The added words to this sentence provide clarification as to what concurrent losses of equipment do not need to be assumed in determining a loss of safety function. The corresponding change to the TS Bases section for LCO 3.0.6 provides additional direction that if an offsite circuit or diesel generator becomes inoperable, then adequate direction is contained in those specific Conditions and Required Actions and that LCO 3.0.6 should not be utilized in those instances.

The second change will add a paragraph at the end of TS 5.5.15. This paragraph will state, " When a loss of safety function is caused by the inoperability of a single Technical Specification support system, the appropriate Conditions and Required Actions to enter are those of the support system." The corresponding change to the TS Bases section for LCO 3.0.6 provides additional direction and clarification on how to apply the concept of a loss of safety function due to the inoperability of a single support system. This proposed change, along with changing Technical Specification Bases for LCO 3.0.6, is consistent with NRC approved TSTF 273-revision 2, as amended by editorial change WOG-ED-23.

For informational purposes, Enclosure 2 provides PVNGS proposed change to LCO 3.0.6 Technical Specification Bases.

TS 5.6.5, COLR. In a continuing effort to improve Accident/Transient Analyses methods, CE Combustion Engineering (ABB-CE) developed the Combustion Engineering Nuclear Transient Simulation (CENTS) computer code. PVNGS proposes to add the option to use the CENTS computer code in licensing analysis as an alternative to the CESEC computer code. CENTS will be added to the list of approved core operating limit analytical methods contained in TS 5.6.5.b. PVNGS intends to replace the presently used transient analysis code CESEC, with CENTS starting with Unit 2, operating cycle 11. CENTS is a best-estimate code designed to provide a realistic simulation of Nuclear Steam Supply System (NSSS) behavior during normal and transient conditions. The CENTS Safety Evaluation (SE) (reference 1) documents the generic NRC approval of the CENTS code for use in the licensing analyses for PWRs designed by CE. The SE stated 5 limitations on the use of the CENTS code:

- The CENTS DNBR calculation for determining overall trends in thermal margin should not be used for licensing analyses
- The application of CENTS is limited to CE plants until additional information is submitted and approved.
- CENTS should not be used for performing Loss of Coolant Accident (LOCA) or severe accident licensing analyses.
- CENTS must only use the point kinetics model in licensing applications.
- CENTS is not approved for performing CEA ejection licensing analyses.
  The NRC SE states that the CEA ejection analyses should be performed with the NRC approved ABB-CE methods of CENPD-190-A (C-E Method for Control Element Assembly Ejection Analysis). However, PVNGS substituted the code described in the reactor coolant pressure calculation, step 8 of CENPD-190-A, CEFLASH-4AS, with CESEC. This is described in the Combustion Engineering Standard Safety Analysis Report (CESSAR) analysis of the CEA ejection event which used CESEC to determine the NSSS response. PVNGS intends to use CENTS in place of CESEC for this application as described below.

Per the NRC safety evaluation, CENTS is not approved for performing CEA ejection licensing analyses. However, the NRC SE states, "The CEA withdrawal event provides the most severe power and heat flux transient for this event classification [reactivity and power distribution anomaly events]. The comparisons for the CEA withdrawal from subcritical and from hot-zero-power provided in Appendices B and C, respectively, indicate good agreement in the predicted power and heat flux transients relative to the predictions with the approved CESEC methodology."

Thus, the CEA withdrawal comparisons should provide sufficient information to validate the CENTS NSSS pressure response for other events in the reactivity and power distribution anomalies category, such as CEA ejection, since they show comparisons of the event with the most rapid NSSS pressure response of these events.

The last paragraph on page 6 of Supplement 1 to CE-NPD 282-P-A, states: "The CESSAR analysis of the CEA ejection event used CESEC to determine the NSSS response. However, this portion of the analysis is not limiting. The STRIKIN-II program was used to calculate the hot rod fuel temperature, heat flux, and the number of fuel rods which experience DNB. ABB-CE methods for the CEA ejection analysis are described in Reference 3.1 [C-E Method for Control Element Assembly Ejection Analysis, CENPD-190-A, January, 1976]."

However, the methods of CENPD-190-A were modified by CESSAR usage of CESEC for the overall NSSS response. In retrospect, this should have been more clearly stated in the above section of Supplement 1 to CENPD 282-P-A.

It is our understanding from reading the safety evaluation, along with Supplement 1 to CE-NPD 282-P-A and the above clarification, that the CEA ejection licensing analysis limitation for CENTS use does not apply to the NSSS response calculation for this event. Thus, the limitation applies only to the calculations that currently employ the STRIKIN-II program.

Also, Palo Verde has performed plant specific benchmark comparison calculations for the CEA ejection event NSSS response using the CESEC and CENTS codes. This portion of the PVNGS CENTS benchmark is attached to this submittal (Enclosure 3). The results generally show good agreement between the two codes with CENTS calculating a slightly lower reactor coolant system pressure. This is explained in Enclosure 3. Therefore, PVNGS intends to use CENTS in place of CESEC to determine the NSSS response to CEA ejection but will not use CENTS for the calculations that currently employ the STRIKIN-II program. This approach is consistent with the current PVNGS licensing basis and with the intent of the limitation in the NRC safety evaluation.

The CENTS code will be used taking into account the five limitations stated above. This change does not immediately alter any methodology used in licensing analysis. It only provides the option to use the CENTS code in place of the CESEC code. The NRC acceptance of the CENTS computer code is described in reference 1. PVNGS intends to implement CENTS in future Palo Verde licensing analyses by following the guidelines prescribed in Generic Letter (GL) 83-11, Supplement 1. The features of GL 83-11, Supplement 1 implementation include the following items:

- Informing the NRC that the guidelines of GL 83-11, Supplement 1, have been followed at least three months before the date of its intended first licensing application.
- Establishing and implementing in-house application procedures, which ensure that the use of approved methods is consistent with the code qualification and, in most instances, with the approved application of the methodology.
- Providing training by either the developer of the code or method, or someone who has been previously qualified in the use of the code or method.
- Verifying the ability to use the methods by comparing their calculated results to an appropriate set of benchmark data, such as physics startup tests, measured

flux detector data during an operating cycle, higher order codes, published numerical benchmarks, analyses of record, etc.

- Conducting safety-related licensing calculations using NRC-approved codes and methods under the control of a Quality Assurance (QA) program which complies with the requirements of Appendix B to 10 CFR 50. This program will also include provisions for evaluating vendor (or other code developer) updates and implementing those updates, if applicable, in codes, methods, and procedures. Additionally, this program will include a provision for informing vendors (or code developers) of any problems or errors discovered while using their codes, methods, or procedures.

When the PVNGS GL 83-11 documentation to implement CENTS is in place, a letter will be sent to the NRC providing a minimum of three months notification prior to the date of PVNGS' intended first licensing application.

The format in TS 5.6.5, Core Operating Limits Report (COLR) to identify the topical report(s) by number, title, date, and NRC staff approval document, will be revised to identify the reports by number and title only. A note will be added to TS 5.6.5.b to specify a complete citation be included in the COLR for each report, including the report number, title, revision, date, and any supplements.

This change has previously been reviewed and accepted by the NRC in letter, "Acceptance for Siemens References to Approved Topical Reports in Technical Specifications" from S.A. Richards, NRC to J.F. Mallay, Siemens Power Corporation dated December 15, 1999. This change is also consistent with NRC accepted TSTF 363-revision 0.

This method of referencing topical reports would allow licensees to use current topical reports to support limits in the COLR without having to submit an amendment to the facility-operating license every time the topical report is revised. The COLR will provide specific information identifying the particular approved topical reports used to determine the core limits in the COLR report. This would eliminate unnecessary expenditure of NRC and licensee resources, and would ease the burden of TS submittal and approval needed to license reload fuel.

D. SAFETY ANALYSIS FOR THE PROPOSED TECHNICAL SPECIFICATION AMENDMENT

TS 5.5.13, Diesel Fuel Oil Testing Program. TS 5.5.13.a.3 will be changed to state; "Water and sediment within limits when tested in accordance with ASTM D1796." This statement is applicable to the water and sediment content of new diesel fuel oil prior to its addition to storage tanks. This proposed amendment change is administrative in nature and has no impact upon the Diesel Fuel Oil Testing Program. The actual limit for diesel fuel oil water and sediment is contained in the TS Bases. This change will correct the misstatement that ASTM D1796 contains the acceptance criteria for "water and sediment" of new diesel fuel oil. ASTM D1796 contains the testing method for "water and sediment" for fuel diesel fuel oil. This proposed change has no affect on the design, operation, or maintenance of PVNGS.

TS 5.5.14, TS Bases Control Program, requires a program for processing changes to the Bases of the Technical Specifications. Based on the changes to 10 CFR 50.59 published in the Federal Register (Volume 64, Number 191) dated October 4, 1999, this specification will be updated. This will entail two changes. The first change will be to the initial sentence for TS 5.5.14.b. The word "involve" will be replaced with "require". Additionally, the second allowance for changing TS Bases as described in TS 5.5.14.b needs to be revised to state, "A change to the updated FSAR or Bases that requires NRC approval pursuant to 10 CFR 50.59." This is consistent with NRC approved TSTF 364-revision 0. This proposed change will allow PVNGS to comply with the approved changes to the 10 CFR 50.59 rule that become effective March 31, 2001. These changes are editorial and reflect changes already reviewed and approved by the NRC and, as such, do not impact the safety of the facility.

TS 5.5.15, SFDP, along with TS Bases for LCO 3.0.6, are revised to clarify the appropriate LCO to enter for loss of function. In addition, the SFDP is revised to state that consideration does not have to be made for a loss of power in determining loss of function. This change does not affect the design, operation, or maintenance of PVNGS but only adds clarification for determining loss of function and for the appropriate LCO(s) to be entered when function is lost. This change is consistent with NRC approved TSTF 273-revision 2, as amended by editorial change WOG-ED-23.

TS 5.6.5, COLR, is being revised to add the option to use the CENTS computer code in licensing analysis by adding CENTS to the list of approved core operating limit analytical methods contained in TS 5.6.5.b. The proposed change will not affect reload analysis other than providing the option to replace the CESEC transient simulation code with an equivalent code. The CENTS code will be used to perform non-LOCA transient design calculations. The CENTS code has been generically approved for the calculation of transient behavior in PWRs designed by Combustion Engineering (CE), subject to certain limitations.

The NRC acceptance of the CENTS computer code is described in letter, "Acceptance for Referencing of Licensing Topical Report CE-NPD 282-P, 'Technical Manual for the CENTS Code'" dated March 17, 1994, from USNRC to S.A. Toelle, ABB Combustion Engineering. PVNGS intends to implement CENTS in future Palo Verde licensing analyses by following the guidelines prescribed in Generic Letter (GL) 83-11, Supplement 1. Since the CENTS code will be used for safety analyses for which it was approved, the proposed change does not impact the safety of the facility.

TS 5.6.5, COLR, will be revised from the current method of identifying the topical report(s) by number, title, date, and NRC staff approval document to identifying the reports by number and title only. A note will be added to TS 5.6.5.b to specify a complete citation be included in the COLR for each report including the report number, title, revision, date, and any supplements.

This change has previously been reviewed and accepted by the NRC in letter, "Acceptance for Siemens References to Approved Topical Reports in Technical Specifications" from S.A. Richards, NRC to J.F. Mallay, Siemens Power Corporation dated December 15, 1999. This change is also consistent with NRC accepted TSTF 363-revision 0.

E. <u>NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION</u>

The Commission has provided standards for determining whether a significant hazards consideration exists as stated in 10 CFR 50.92. A proposed amendment to an operating license for a facility involves no significant hazards consideration if operation of the facility in accordance with a proposed amendment would not: (1) Involve a significant increase in the probability or consequences of an accident previously evaluated; or (2) Create the possibility of a new or different kind of accident from any accident previously evaluated; or (3) Involve a significant reduction in a margin of safety. A discussion of these standards as they relate to this amendment request follows:

<u>Standard 1</u> – Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?

No. The proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

Technical Specification (TS) 5.5.13, Diesel Fuel Oil Testing Program. TS 5.5.13.a.3 currently states, "Water and sediment are within the limits of ASTM D1796," for the acceptability of new diesel fuel oil. This is an incorrect reference for the limits of water and sediment content of new fuel oil. The water and sediment limits for new fuel oil are contained within the Technical Specification Bases. ASTM D1796 contains testing methods used for analysis of new fuel oil for water and sediment. This proposed amendment changes the wording of TS 5.5.13.a.3 to state, "Water and sediment within limits when tested in accordance with ASTM D1796." This proposed change is an administrative change and will have no affect on plant design, operation, or maintenance. Additionally, this proposed change does not result in any hardware changes or affect plant operating practices. The water and sediment testing methods and limits are not affected by this change. Thus, this proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

TS 5.5.14, TS Bases Control Program, requires a program for processing changes to the Bases of the TS. TS 5.5.14.b states,

> "Licensees may make changes to Bases without prior NRC approval provided the changes do not involve either of the following:
>
> > A change in the TS incorporated in the license; or
> >
> > A change to the updated FSAR or Bases that involves an unreviewed safety question as defined in 10 CFR 50.59."

In the initial sentence to TS 5.5.14.b, the word "involve" will be replaced with "require". Additionally, the second allowance for changing TS Bases as described in TS 5.5.14.b will be revised to state, "A change to the updated FSAR or Bases that requires NRC approval pursuant to 10 CFR 50.59." This change is based on the changes to 10 CFR 50.59 published in the Federal Register (Volume 64, Number 191) dated October 4, 1999. This change is consistent with NRC approved Technical Specifications Task Force (TSTF) traveler number 364-revision 0.

This change will also numerically format the two options listed in TS 5.5.14.b. This is consistent with other listings contained in Section 5.0 of the TS.

This proposed change deletes the reference to "unreviewed safety question" as previously used in 10 CFR 50.59. Deletion of this definition was approved by the NRC with the revision to 10 CFR 50.59. This proposed change is an administrative change and will have no affect on plant design, operation, or maintenance. Additionally, this proposed change does not result in any hardware changes or affect plant operating practices. Therefore, this proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

TS 5.5.15, Safety Functions Determination Program (SFDP). Clarification is being added to TS 5.5.15. The second paragraph of TS 5.5.15 will be changed to read: "A loss of safety function exists when, assuming no concurrent single failure, no concurrent loss of offsite power, or no concurrent loss of onsite diesel generator(s), a safety function assumed in the accident analysis cannot be performed. For the purpose of this program, a loss of safety function may exist when a support system is inoperable, and: ....."

An additional paragraph will be added to the end of TS 5.5.15 stating, "When a loss of safety function is caused by the inoperability of a single Technical Specification support system, the appropriate Conditions and Required Actions to enter are those of the support system."

Additionally, clarification will be added to Limiting Condition for Operation (LCO) 3.0.6 Bases of the "appropriate LCO for loss of safety function." The Bases will also clarify the requirements for the SFDP that consideration does not have to be made for a loss of power in determining loss of function. This change is consistent with NRC approved TSTF traveler number 273-revision 2, as amended by editorial change WOG-ED-23.

In addition, an editorial change to remove the "s" from the word "Functions" in the title for TS 5.5.15 will occur. This change reflects the plant specific name for this program.

This proposed change is an administrative change and will have no affect on plant design, operation, or maintenance. The change clarifies the requirements for determining loss of safety function and the correct LCO to enter for loss of safety function. The proposed change does not result in any hardware changes or affect plant operating practices. The program will still determine when a safety function has been lost and will direct the appropriate actions. Therefore, this proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

TS 5.6.5, Core Operating Limits Report (COLR) is being revised to add the option to use the CENTS computer code in licensing analysis by adding CENTS to the list of approved core operating limit analytical methods contained in TS 5.6.5.b. The CENTS computer code has been generically approved for the calculation of transient behavior in Pressurized Water Reactors (PWRs) designed by Combustion Engineering (CE). PVNGS intends to qualify CENTS for use in future Palo Verde licensing analyses by following the guidelines prescribed in Generic Letter (GL) 83-11, Supplement 1.

CENTS is a best-estimate code designed to provide a realistic simulation of Nuclear Steam Supply System (NSSS) behavior during normal and transient conditions. The CENTS Safety Evaluation (SE) documents the generic NRC approval of the CENTS code for use in the licensing analyses for PWRs designed by CE. The CENTS SE is described in letter, "Acceptance for Referencing of Licensing Topical Report CE-NPD 282-P, 'Technical Manual for the CENTS Code'" dated March 17, 1994, from USNRC to S.A. Toelle, ABB Combustion Engineering.

The proposed change does not immediately alter any methodology used in reload analysis. It only provides the option to replace the CESEC transient simulation code with an alternate NRC approved code. Providing the option to substitute the NRC approved CESEC code with another NRC approved code (CENTS) will not alter the physical characteristics of any component involved in the initiation or mitigation of an accident. The actual implementation of the CENTS code will be performed by following the guidance provided in Generic Letter (GL) 83-11, Supplement 1. This proposed change does not result in any hardware changes or affect plant operating practices. Thus, this proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

TS 5.6.5, Core Operating Limits Report (COLR) which identifies the methodology report(s) by number, title, date, and NRC staff approval document, will be revised to allow the reports to be identified by number and title only. A note will be added to TS 5.6.5.b to specify that a complete citation be included in the COLR for each report, including the report number, title, revision, date, and any supplements.

This change has previously been reviewed and accepted by the NRC in letter, "Acceptance for Siemens References to Approved Topical Reports in Technical Specifications" from S.A. Richards, NRC to J.F. Mallay, Siemens Power Corporation dated December 15, 1999. This change is also consistent with NRC accepted TSTF 363-revision 0.

Additionally, TS 5.6.5.b.6 and 5.6.5.b.7 both list the same topical report (Calculative Methods for the CE Small Break LOCA Evaluation Model, CENPD-137). TS 5.6.5.b.7 is the supplement to the topical report listed in 5.6.5.b.6. TS 5.6.5.b.7 will be deleted and the "Calculative Methods for the CE Small Break LOCA Evaluation Model, CENPD-137" topical report (along with its supplement) will be listed in full text within the COLR.

This proposed change is an administrative change and will have no affect on plant design, operation, or maintenance. Thus, this proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

Standard 2 -- Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?

No. The proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

TS 5.5.13, Diesel Fuel Oil Testing Program. The proposed change to TS 5.5.13.a.3 is an administrative change. This change would have no affect on the physical plant. Consequently, plant configuration and the operational characteristics remain unchanged and the proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

TS 5.5.14, TS Bases Control Program. The proposed changes associated with TS 5.5.14.b do not involve any physical changes. These changes allow PVNGS to be in compliance with NRC approved changes to 10 CFR 50.59. This change is an administrative change. Plant configuration and the operational characteristics remain unchanged and thus, the proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

TS 5.5.15, SFDP. The proposed change to TS 5.5.15 does not involve any physical changes to the plant. This change is an administrative change. The loss of function of the specific component is addressed in its specific TS LCO and plant configuration will be governed by the required actions of those LCOs. Since this proposed change is a clarification that does not degrade the availability or capability of safety related equipment, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

TS 5.6.5, COLR is being revised to add the option to use the CENTS computer code in licensing analysis by adding CENTS to the list of approved core operating limit analytical methods contained in TS 5.6.5.b. The proposed change will not affect reload analysis other than providing an option to replace the CESEC transient simulation code with an equivalent code. Providing this option in and of itself will not alter the physical characteristics of any component in the plant. Since providing the option to use the CENTS code will not alter the physical characteristics of any component in the plant, this proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

TS 5.6.5, Core Operating Limits Report (COLR) which identifies the methodology report(s) by number, title, date, and NRC staff approval document, will be revised to allow the reports to be identified by number and title only. This is an administrative change. This change has no affect on the physical plant. Plant configuration and the operational characteristics remain unchanged and thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

Standard 3 -- Does the proposed change involve a significant reduction in a margin of safety?

No. The proposed change does not involve a significant reduction in a margin of safety.

TS 5.5.13, Diesel Fuel Oil Testing Program. The proposed change to TS 5.5.13.a.3 is an administrative change. This change would have no affect on the physical plant and has no affect on any safety analyses assumptions. Therefore, this proposed change does not involve a significant reduction in a margin of safety.

TS 5.5.14, TS Bases Control Program. The proposed change to TS 5.5.14.b will not reduce a margin of safety because it has no direct effect on any safety analyses assumptions. Changes to the TS Bases that result in meeting the criteria in paragraph (c)(2) of 10 CFR 50.59 will still require NRC approval pursuant to 10 CFR 50.59. This change is administrative in nature and is based on NRC reviewed and approved changes to 10 CFR 50.59. Therefore, this proposed change does not involve a significant reduction in a margin of safety.

TS 5.5.15, SFDP. The proposed changes to TS 5.5.15 are clarifications only. No changes are made in the LCO, the time required for the TS required actions to be completed, or the out of service time for the components involved. The NRC has approved the proposed administrative changes (TSTF 273-revision 2, as amended by editorial change WOG-ED-23). Safety-related equipment controlled by the TS will still perform as credited in the safety analysis. Therefore, this proposed change does not involve a significant reduction in a margin of safety.

TS 5.6.5, COLR is being revised to add the option to use the CENTS computer code in licensing analysis by adding CENTS to the list of approved core operating limit analytical methods. This proposed change will allow running existing analyses with a different method that has been reviewed and approved by the NRC. The actual implementation of the CENTS code will be performed by following the guidance provided in Generic Letter (GL) 83-11, Supplement 1. Thus, this proposed change does not involve a significant reduction in a margin of safety.

TS 5.6.5, Core Operating Limits Report (COLR) which identifies the methodology report(s) by number, title, date, and NRC staff approval document, will be revised to allow the reports to be identified by number and title only. This is an administrative change. This change has no affect on the physical plant. Plant configuration and the operational characteristics remain unchanged. Therefore, this change does not involve a significant reduction in a margin of safety.

Based on the responses to these three criteria, Arizona Public Service Company (APS) has concluded that the proposed amendment involves no significant hazards consideration.

F. <u>ENVIRONMENTAL CONSIDERATION</u>

APS has determined that the proposed amendment involves no changes in the amount or type of effluent that may be released offsite, and results in no increase in individual or cumulative occupational radiation exposure. As described above, the proposed

Technical Specification amendment involves no significant hazards consideration and, as such, meets the eligibility criteria for categorical exclusion set forth in 10CFR 51.22(c)(9).

Reference:

1. Acceptance for Referencing of Licensing Topical Report CE-NPD 282-P, "Technical Manual for the CENTS Code", from Martin J. Virgilio, NRC to SA Toelle, ABB Combustion Engineering, dated 3/17/94

G. MARKED-UP TECHNICAL SPECIFICATION PAGES

Units 1, 2, and 3;	Pages 5.5-21 through 5.5-24
	Page 5.6-3 through 5.6-6

5.5 Programs and Manuals (continued)

5.5.13 Diesel Fuel Oil Testing Program (continued)

a. Acceptability of new fuel oil for use prior to addition to storage tanks by determining that the fuel oil has:

1. An API gravity or an absolute specific gravity within limits,

2. A flash point and kinematic viscosity within limits for ASTM 2D fuel oil, and

3. Water and sediment ~~are~~ within ~~the~~ limits **when tested in accordance with** ~~of~~ ASTM D1796.

b. Other properties for ASTM 2D fuel oil are within limits within 31 days following sampling and addition to storage tanks; and

c. Total particulate concentration of the stored fuel oil is ≤ 10 mg/l when tested every 92 days in accordance with ASTM D-2276, Method A-2 or A-3.

5.5.14 Technical Specifications (TS) Bases Control Program

This program provides a means for processing changes to the Bases of these Technical Specifications.

a. Changes to the Bases of the TS shall be made under appropriate administrative controls and reviews.

b. Licensees may make changes to Bases without prior NRC approval provided the changes do not ~~involve~~ **require** either of the following:

1. A change in the TS incorporated in the license; or

2. A change to the updated FSAR or Bases that ~~involves an unreviewed safety question as defined in 10 CFR 50.59~~. **requires NRC approval pursuant to 10 CFR 50.59.**

c. The Bases Control Program shall contain provisions to ensure that the Bases are maintained consistent with the FSAR.

(continued)

5.5 Programs and Manuals (continued)

5.5.14 Technical Specifications (TS) Bases Control Program (continued)

d. Proposed changes that meet the criteria of Specification 5.5.14b above shall be reviewed and approved by the NRC prior to implementation. Changes to the Bases implemented without prior NRC approval shall be provided to the NRC on a frequency consistent with 10 CFR 50.71(e).

5.5.15 Safety Function~~s~~ Determination Program (SFDP)

This program ensures loss of safety function is detected and appropriate actions taken. Upon entry into LCO 3.0.6, an evaluation shall be made to determine if loss of safety function exists. Additionally, other appropriate limitations and remedial or compensatory actions may be identified to be taken as a result of the support system inoperability and corresponding exception to entering supported system Condition and Required Actions. This program implements the requirements of LCO 3.0.6. The SFDP shall contain the following:

a. Provisions for cross train checks to ensure a loss of the capability to perform the safety function assumed in the accident analysis does not go undetected;

b. Provisions for ensuring the plant is maintained in a safe condition if a loss of function condition exists;

c. Provisions to ensure that an inoperable supported system's Completion Time is not inappropriately extended as a result of multiple support system inoperabilities; and

d. Other appropriate limitations and remedial or compensatory actions.

A loss of safety function exists when, assuming no concurrent single failure, **no concurrent loss of offsite power, or no concurrent loss of onsite diesel generator(s),** a safety function assumed in the accident analysis cannot be performed. For the purpose of this program, a loss of safety function may exist when a support system is inoperable, and:

a. A required system redundant to system(s) supported by the inoperable support system is also inoperable; or

(continued)

5.5 Programs and Manuals (continued)

5.5.15 Safety ~~Functions~~ Determination Program (continued)

b. A required system redundant to system(s) in turn supported by the inoperable supported system is also inoperable; or

c. A required system redundant to support system(s) for the supported systems (a) and (b) above is also inoperable.

The SFDP identifies where a loss of safety function exists. If a loss of safety function is determined to exist by this program, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered.

When a loss of safety function is caused by the inoperability of a single Technical Specification support system, the appropriate Conditions and Required Actions to enter are those of the support system.

5.5.16 Containment Leakage Rate Testing Program

A program shall be established to implement the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," dated September, 1995, as modified by the following exceptions:

The peak calculated containment internal pressure for the design basis loss of coolant accident, P_a, is 52.0 psig. The containment design pressure is 60 psig.

The maximum allowable containment leakage rate, L_a, at P_a, shall be 0.1 % of containment air weight per day.

Leakage Rate acceptance criteria are:

a. Containment leakage rate acceptance criterion is $\leq 1.0\ L_a$. During the first unit startup following testing in accordance with this program, the leakage rate acceptance are $< 0.60\ L_a$ for the Type B and C tests and $\leq 0.75\ L_a$ for Type A tests.

b. Air lock testing acceptance criteria are:

1. Overall air lock leakage rate is $\leq 0.05\ L_a$ when tested at $\geq P_a$

(continued)

5.5 Programs and Manuals (continued)

5.5.16 Containment Leakage Rate Testing Program (continued)

2. For each door, leakage rate is ≤ 0.01 L_a when pressurized to ≥ 14.5 psig.

The provisions of SR 3.0.2 do not apply to the test frequencies in the Containment Leakage Rate Testing Program.

The provisions of SR 3.0.3 are applicable to the Containment Leakage Rate Testing Program.

(continued)

5.6 Reporting Requirements (continued)

5.6.5 CORE OPERATING LIMITS REPORT (COLR)

a. Core operating limits shall be established prior to each reload cycle, or prior to any remaining portion of a reload cycle, and shall be documented in the COLR for the following:

1. Shutdown Margin - Reactor Trip Breakers Open for Specification 3.1.1.
2. Shutdown Margin - Reactor Trip Breakers Closed for Specification 3.1.2.
3. Moderator Temperature Coefficient BOL and EOL limits for Specification 3.1.4.
4. Boron Dilution Alarm System for Specification 3.3.12.
5. CEA Alignment for Specification 3.1.5.
6. Regulating CEA Insertion Limits for Specification 3.1.7.
7. Part Length CEA Insertion Limits for Specification 3.1.8.
8. Linear Heat Rate for Specification 3.2.1.
9. Azimuthal Power Tilt - T_q for Specification 3.2.3.
10. DNBR for Specification 3.2.4.
11. Axial Shape Index for Specification 3.2.5.
12. Boron Concentration (Mode 6) for Specification 3.9.1.

b. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents:

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The COLR will contain the complete identification for each of the Technical Specification referenced topical reports used to prepare the COLR (i.e., report number, title, revision, date, and any supplements).

(continued)

5.6 Reporting Requirements (continued)

5.6.5 Core Operating Limits Report (COLR) (continued)

1. "CE Method for Control Element Assembly Ejection Analysis, "CENPD-0190-A, ~~January 1976~~ (Methodology for Specification 3.1.7, Regulating CEA Insertion Limits).

2. "The ROCS and DIT Computer Codes for Nuclear Design," CENPD-266-P-A, ~~April 1983~~ [Methodology for Specifications 3.1.1, Shutdown Margin - Reactor Trip Breakers Open; 3.1.2, Shutdown Margin - Reactor Trip Breakers Closed; 3.1.4, Moderator Temperature Coefficient BOL and EOL limits; 3.1.7, Regulating CEA Insertion Limits and 3.9.1, Boron Concentration (Mode 6)].

3. "Safety Evaluation Report related to the Final Design of the Standard Nuclear Steam Supply Reference Systems CESSAR System 80, Docket No. STN 50-470, "NUREG-0852 (November 1981), Supplements No. 1 (March 1983), No. 2 (September 1983), No. 3 (December 1987) [Methodology for Specifications 3.1.2, Shutdown Margin - Reactor Trip Breakers Closed; 3.1.4, Moderator Temperature Coefficient BOL and EOL limits; 3.3.12, Boron Dilution Alarm System; 3.1.5, CEA Alignment; 3.1.7, Regulating CEA Insertion Limits; 3.1.8, Part Length CEA Insertion Limits and 3.2.3, Azimuthal Power Tilt - T_q].

4. "Modified Statistical Combination of Uncertainties," CEN-356(V)-P-A ~~Revision 01-P-A, May 1988~~ and "System 80™ Inlet Flow Distribution," Supplement 1-P to Enclosure 1-P to LD-82-054, ~~February 1993~~ (Methodology for Specification 3.2.4, DNBR and 3.2.5 Axial Shape Index).

5. "Calculative Methods for the CE Large Break LOCA Evaluation Model ~~for the Analysis of CE and W Designed NSSS,~~" CENPD-132, ~~Supplement 3-P-A, June 1985~~ (Methodology for Specification 3.2.1, Linear Heat Rate).

6. "Calculative Methods for the CE Small Break LOCA Evaluation Model," CENPD-137-P, ~~August 1974~~ (Methodology for Specification 3.2.1, Linear Heat Rate).

~~7. "Calculative Methods for the CE Small Break LOCA Evaluation Model," CENPD 137 P, Supplement 1P, January 1977 (Methodology for Specification 3.2.1, Linear Heat Rate).~~

(continued)

5.6 Reporting Requirements (continued)

5.6.5 Core Operating Limits Report (COLR) (continued)

7. ~~8~~. Letter: O.D. Parr (NRC) to F. M. Stern (CE), dated June 13, 1975 (NRC Staff Review of the Combustion Engineering ECCS Evaluation Model). NRC approval for: 5.6.5.b.6.

8. ~~9~~. Letter: K. Kniel (NRC) to A. E. Scherer (CE), dated September 27, 1977 (Evaluation of Topical Reports CENPD-133, Supplement 3-P and CENPD-137, Supplement 1-P). NRC approval for 5.6.5.b.7.

9. ~~10~~. "Fuel Rod Maximum Allowable Pressure," CEN-372-P-A, ~~May 1990~~ (Methodology for Specification 3.2.1, Linear Heat Rate).

10. ~~11~~. Letter: A. C. Thadani (NRC) to A. E. Scherer (CE), dated April 10, 1990, ("Acceptance for Reference CE Topical Report CEN-372-P"). NRC approval for 5.6.5.b.10.

11. ~~12~~. "Arizona Public Service Company PWR Reactor Physics Methodology Using CASMO-4/SIMULATE-3," ~~September 1999~~ [Methodology for Specifications 3.1.1, Shutdown Margin - Reactor Trip Breakers Open; 3.1.2, Shutdown Margin - Reactor Trip Breakers Closed; 3.1.4, Moderator Temperature Coefficient; 3.1.7, Regulating CEA Insertion Limits and 3.9.1, Boron Concentration (Mode 6)].

12. **"Technical Manual for the CENTS Code," CE-NPD 282-P-A, Volumes 1-3, [Methodology for Specifications 3.1.2, Shutdown Margin-Reactor Trip Breakers Closed; 3.1.4, Moderator Temperature Coefficient; 3.1.5, CEA Alignment; 3.1.7, Regulating CEA Insertion Limits; 3.1.8, Part Length CEA Insertion Limits and 3.2.3, Azimuthal Power Tilt- T_q].**

c. The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal mechanical limits, core thermal hydraulic limits, Emergency Core Cooling Systems (ECCS) limits, nuclear limits such as SDM, transient analysis limits, and accident analysis limits) of the safety analysis are met.

d. The COLR, including any mid cycle revisions or supplements, shall be provided upon issuance for each reload cycle to the NRC.

5.6 Reporting Requirements (continued)

5.6.6 PAM Report

When a report is required by Condition B or G of LCO 3.3.10, "Post Accident Monitoring (PAM) Instrumentation," a report shall be submitted within the following 14 days. The report shall outline the preplanned alternate method of monitoring, the cause of the inoperability, and the plans and schedule for restoring the instrumentation channels of the Function to OPERABLE status.

5.6.7 Tendon Surveillance Report

Any abnormal degradation of the containment structure detected during the tests required by the Pre-Stressed Concrete Containment Tendon Surveillance Program shall be reported to the NRC within 30 days. The report shall include a description of the tendon condition, the condition of the concrete (especially at tendon anchorages), the inspection procedures, the tolerances on cracking, and the corrective action taken.

5.6.8 Steam Generator Tube Inspection Report

Within 15 days following the completion of each inservice inspection of steam generator tubes, the number of tubes plugged and/or repaired in each steam generator shall be reported to the Commission in a Special Report.

The complete results of the steam generator tube inservice inspection shall be submitted to the Commission in a Special Report within 12 months following completion of the inspection. This Special Report shall include:

a. Number and extent of tubes inspected.

b. Location and percent of wall-thickness penetration for each indication of an imperfection.

c. Identification of tubes plugged and/or repaired.

Results of steam generator tube and sleeve inspections which fall into Category C-3 shall be reported in a Special Report to the Commission within 30 days and prior to resumption of plant operation and shall provide a description of investigations conducted to determine cause of the tube degradation and corrective measures taken to prevent recurrence.

G. RETYPED TECHNICAL SPECIFICATION PAGES

Units 1, 2, and 3;	**Pages 5.5-21 through 5.5-24** **Pages 5.6-3 through 5.6-6**

5.5 Programs and Manuals (continued)

5.5.13 Diesel Fuel Oil Testing Program (continued)

a. Acceptability of new fuel oil for use prior to addition to storage tanks by determining that the fuel oil has:

   1. An API gravity or an absolute specific gravity within limits,

   2. A flash point and kinematic viscosity within limits for ASTM 2D fuel oil, and

   3. Water and sediment within limits when tested in accordance with ASTM D1796;

b. Other properties for ASTM 2D fuel oil are within limits within 31 days following sampling and addition to storage tanks; and

c. Total particulate concentration of the stored fuel oil is ≤ 10 mg/l when tested every 92 days in accordance with ASTM D-2276, Method A-2 or A-3.

5.5.14 Technical Specifications (TS) Bases Control Program

This program provides a means for processing changes to the Bases of these Technical Specifications.

a. Changes to the Bases of the TS shall be made under appropriate administrative controls and reviews.

b. Licensees may make changes to Bases without prior NRC approval provided the changes do not require either of the following:

   1. A change in the TS incorporated in the license; or

   2. A change to the updated FSAR or Bases that requires NRC approval pursuant to 10 CFR 50.59.

c. The Bases Control Program shall contain provisions to ensure that the Bases are maintained consistent with the FSAR.

(continued)

5.5 Programs and Manuals (continued)

5.5.14 Technical Specifications (TS) Bases Control Program (continued)

d. Proposed changes that meet the criteria of Specification 5.5.14b above shall be reviewed and approved by the NRC prior to implementation. Changes to the Bases implemented without prior NRC approval shall be provided to the NRC on a frequency consistent with 10 CFR 50.71(e).

5.5.15 Safety Function Determination Program (SFDP)

This program ensures loss of safety function is detected and appropriate actions taken. Upon entry into LCO 3.0.6, an evaluation shall be made to determine if loss of safety function exists. Additionally, other appropriate limitations and remedial or compensatory actions may be identified to be taken as a result of the support system inoperability and corresponding exception to entering supported system Condition and Required Actions. This program implements the requirements of LCO 3.0.6. The SFDP shall contain the following:

a. Provisions for cross train checks to ensure a loss of the capability to perform the safety function assumed in the accident analysis does not go undetected;

b. Provisions for ensuring the plant is maintained in a safe condition if a loss of function condition exists;

c. Provisions to ensure that an inoperable supported system's Completion Time is not inappropriately extended as a result of multiple support system inoperabilities; and

d. Other appropriate limitations and remedial or compensatory actions.

A loss of safety function exists when, assuming no concurrent single failure, no concurrent loss of offsite power, or no concurrent loss of onsite diesel generator(s), a safety function assumed in the accident analysis cannot be performed. For the purpose of this program, a loss of safety function may exist when a support system is inoperable, and:

a. A required system redundant to system(s) supported by the inoperable support system is also inoperable; or

(continued)

5.5 Programs and Manuals (continued)

5.5.15 Safety Function Determination Program (continued)

b. A required system redundant to system(s) in turn supported by the inoperable supported system is also inoperable; or

c. A required system redundant to support system(s) for the supported systems (a) and (b) above is also inoperable.

The SFDP identifies where a loss of safety function exists. If a loss of safety function is determined to exist by this program, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered.

When a loss of safety function is caused by the inoperability of a single Technical Specification support system, the appropriate Conditions and Required Actions to enter are those of the support system.

5.5.16 Containment Leakage Rate Testing Program

A program shall be established to implement the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," dated September, 1995, as modified by the following exceptions:

The peak calculated containment internal pressure for the design basis loss of coolant accident, P_a, is 52.0 psig. The containment design pressure is 60 psig.

The maximum allowable containment leakage rate, L_a, at P_a, shall be 0.1 % of containment air weight per day.

Leakage Rate acceptance criteria are:

a. Containment leakage rate acceptance criterion is $\leq 1.0\ L_a$. During the first unit startup following testing in accordance with this program, the leakage rate acceptance are $< 0.60\ L_a$ for the Type B and C tests and $\leq 0.75\ L_a$ for Type A tests.

(continued)

5.5 Programs and Manuals (continued)

5.5.16 Containment Leakage Rate Testing Program (continued)

b. Air lock testing acceptance criteria are:

1. Overall air lock leakage rate is $\leq 0.05\ L_a$ when tested at $\geq P_a$.

2. For each door, leakage rate is $\leq 0.01\ L_a$ when pressurized to $\geq$14.5 psig.

The provisions of SR 3.0.2 do not apply to the test frequencies in the Containment Leakage Rate Testing Program.

The provisions of SR 3.0.3 are applicable to the Containment Leakage Rate Testing Program.

5.6 Reporting Requirements (continued)

5.6.5 CORE OPERATING LIMITS REPORT (COLR)

a. Core operating limits shall be established prior to each reload cycle, or prior to any remaining portion of a reload cycle, and shall be documented in the COLR for the following:

1. Shutdown Margin - Reactor Trip Breakers Open for Specification 3.1.1.
2. Shutdown Margin - Reactor Trip Breakers Closed for Specification 3.1.2.
3. Moderator Temperature Coefficient BOL and EOL limits for Specification 3.1.4.
4. Boron Dilution Alarm System for Specification 3.3.12.
5. CEA Alignment for Specification 3.1.5.
6. Regulating CEA Insertion Limits for Specification 3.1.7.
7. Part Length CEA Insertion Limits for Specification 3.1.8.
8. Linear Heat Rate for Specification 3.2.1.
9. Azimuthal Power Tilt - T_q for Specification 3.2.3.
10. DNBR for Specification 3.2.4.
11. Axial Shape Index for Specification 3.2.5.
12. Boron Concentration (Mode 6) for Specification 3.9.1.

b. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents:

-----------------------NOTE-------------------------

The COLR will contain the complete identification for each of the Technical Specification referenced topical reports used to prepare the COLR (i.e., report number, title, revision, date, and any supplements).

(continued)

5.6 Reporting Requirements (continued)

5.6.5 Core Operating Limits Report (COLR) (continued)

1. "CE Method for Control Element Assembly Ejection Analysis," CENPD-0190-A, (Methodology for Specification 3.1.7, Regulating CEA Insertion Limits).

2. "The ROCS and DIT Computer Codes for Nuclear Design," CENPD-266-P-A, [Methodology for Specifications 3.1.1, Shutdown Margin - Reactor Trip Breakers Open; 3.1.2, Shutdown Margin - Reactor Trip Breakers Closed; 3.1.4, Moderator Temperature Coefficient BOL and EOL limits; 3.1.7, Regulating CEA Insertion Limits and 3.9.1, Boron Concentration (Mode 6)].

3. "Safety Evaluation Report related to the Final Design of the Standard Nuclear Steam Supply Reference Systems CESSAR System 80, Docket No. STN 50-470, "NUREG-0852 (November 1981), Supplements No. 1 (March 1983), No. 2 (September 1983), No. 3 (December 1987) [Methodology for Specifications 3.1.2, Shutdown Margin - Reactor Trip Breakers Closed; 3.1.4, Moderator Temperature Coefficient BOL and EOL limits; 3.3.12, Boron Dilution Alarm System; 3.1.5, CEA Alignment; 3.1.7, Regulating CEA Insertion Limits; 3.1.8, Part Length CEA Insertion Limits and 3.2.3, Azimuthal Power Tilt - T_q].

4. "Modified Statistical Combination of Uncertainties," CEN-356(V)-P-A and "System 80™ Inlet Flow Distribution," Supplement 1-P to Enclosure 1-P to LD-82-054, (Methodology for Specification 3.2.4, DNBR and 3.2.5 Axial Shape Index).

5. "Calculative Methods for the CE Large Break LOCA Evaluation Model," CENPD-132, (Methodology for Specification 3.2.1, Linear Heat Rate).

6. "Calculative Methods for the CE Small Break LOCA Evaluation Model," CENPD-137-P, (Methodology for Specification 3.2.1, Linear Heat Rate).

(continued)

5.6 Reporting Requirements (continued)

5.6.5 Core Operating Limits Report (COLR) (continued)

7. Letter: O.D. Parr (NRC) to F. M. Stern (CE), dated June 13, 1975 (NRC Staff Review of the Combustion Engineering ECCS Evaluation Model). NRC approval for: 5.6.5.b.6.

8. Letter: K. Kniel (NRC) to A. E. Scherer (CE), dated September 27, 1977 (Evaluation of Topical Reports CENPD-133, Supplement 3-P and CENPD-137, Supplement 1-P). NRC approval for 5.6.5.b.7.

9. "Fuel Rod Maximum Allowable Pressure," CEN-372-P-A, (Methodology for Specification 3.2.1, Linear Heat Rate).

10. Letter: A. C. Thadani (NRC) to A. E. Scherer (CE), dated April 10, 1990, ("Acceptance for Reference CE Topical Report CEN-372-P"). NRC approval for 5.6.5.b.10.

11. "Arizona Public Service Company PWR Reactor Physics Methodology Using CASMO-4/SIMULATE-3," [Methodology for Specifications 3.1.1, Shutdown Margin - Reactor Trip Breakers Open; 3.1.2, Shutdown Margin - Reactor Trip Breakers Closed; 3.1.4, Moderator Temperature Coefficient; 3.1.7, Regulating CEA Insertion Limits and 3.9.1, Boron Concentration (Mode 6)].

12. "Technical Manual for the CENTS Code," CE-NPD 282-P-A, Volumes 1-3, [Methodology for Specifications 3.1.2, Shutdown Margin-Reactor Trip Breakers Closed; 3.1.4, Moderator Temperature Coefficient; 3.1.5, CEA Alignment; 3.1.7, Regulating CEA Insertion Limits; 3.1.8, Part Length CEA Insertion Limits and 3.2.3, Azimuthal Power Tilt- T_q].

c. The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal mechanical limits, core thermal hydraulic limits, Emergency Core Cooling Systems (ECCS) limits, nuclear limits such as SDM, transient analysis limits, and accident analysis limits) of the safety analysis are met.

d. The COLR, including any mid cycle revisions or supplements, shall be provided upon issuance for each reload cycle to the NRC.

(continued)

5.6 Reporting Requirements (continued)

5.6.6 PAM Report

When a report is required by Condition B or G of LCO 3.3.10, "Post Accident Monitoring (PAM) Instrumentation," a report shall be submitted within the following 14 days. The report shall outline the preplanned alternate method of monitoring, the cause of the inoperability, and the plans and schedule for restoring the instrumentation channels of the Function to OPERABLE status.

5.6.7 Tendon Surveillance Report

Any abnormal degradation of the containment structure detected during the tests required by the Pre-Stressed Concrete Containment Tendon Surveillance Program shall be reported to the NRC within 30 days. The report shall include a description of the tendon condition, the condition of the concrete (especially at tendon anchorages), the inspection procedures, the tolerances on cracking, and the corrective action taken.

5.6.8 Steam Generator Tube Inspection Report

Within 15 days following the completion of each inservice inspection of steam generator tubes, the number of tubes plugged and/or repaired in each steam generator shall be reported to the Commission in a Special Report.

The complete results of the steam generator tube inservice inspection shall be submitted to the Commission in a Special Report within 12 months following completion of the inspection. This Special Report shall include:

a. Number and extent of tubes inspected.

b. Location and percent of wall-thickness penetration for each indication of an imperfection.

c. Identification of tubes plugged and/or repaired.

Results of steam generator tube and sleeve inspections which fall into Category C-3 shall be reported in a Special Report to the Commission within 30 days and prior to resumption of plant operation and shall provide a description of investigations conducted to determine cause of the tube degradation and corrective measures taken to prevent recurrence.

ENCLOSURE 2

Proposed Changes to LCO 3.0.6 Technical Specification Bases (for information only)

BASES

LCO 3.0.6 (continued)

However, there are instances where a support system's Required Action may either direct a supported system to be declared inoperable or direct entry into Conditions and Required Actions for the supported system. This may occur immediately or after some specified delay to perform some other Required Action. Regardless of whether it is immediate or after some delay, when a support system's Required Action directs a supported system to be declared inoperable or directs entry into Conditions and Required Actions for a supported system, the applicable Conditions and Required Actions shall be entered in accordance with LCO 3.0.2.

Specification 5.5.15, "Safety Function Determination Program (SFDP)," ensures loss of safety function is detected and appropriate actions are taken. Upon entry into LCO 3.0.6, an evaluation shall be made to determine if loss of safety function exists. Additionally, other limitations, remedial actions, or compensatory actions may be identified as a result of the support system inoperability and corresponding exception to entering supported system Conditions and Required Actions. The SFDP implements the requirements of LCO 3.0.6.

Cross train checks to identify a loss of safety function for those support systems that support multiple and redundant safety systems are required. The cross train check verifies that the supported systems of the redundant OPERABLE support system are OPERABLE, thereby ensuring safety function is retained. A loss of safety function may exist when a support system is inoperable, and:

a. A required system redundant to system(s) supported by the inoperable support system is also inoperable; or (EXAMPLE B3.0.6-1)

b. A required system redundant to system(s) in turn supported by the inoperable supported system is also inoperable; or (EXAMPLE B3.0.6-2)

c. A required system redundant to support system(s) for the supported systems (a) and (b) above is also inoperable. (EXAMPLE B3.0.6-3)

(continued)

BASES

LCO 3.0.6 (continued)

If this evaluation determines that a loss of safety function exists, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered.

This loss of safety function does not require the assumption of additional single failures or loss of offsite power. Since operation is being restricted in accordance with the ACTIONS of the support system, any resulting temporary loss of redundancy or single failure protection is taken into account. Similarly, the ACTIONS for inoperable offsite circuit(s) and inoperable diesel generator(s) provide the necessary restriction for cross train inoperabilities. This explicit cross train verification for inoperable AC electrical power sources also acknowledges that supported system(s) are not declared inoperable solely as a result of inoperability of a normal or emergency electrical power source (refer to the definition of OPERABILITY).

When a loss of safety function is determined to exist, and the SFDP requires entry into the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists, consideration must be given to the specific type of function affected. Where a loss of function is solely due to a single Technical Specification support system (e.g., loss of automatic start due to inoperable instrumentation, or loss of pump suction source due to low tank level) the appropriate LCO is the LCO for the support system. The ACTIONS for a support system LCO adequately addresses the inoperabilities of that system without reliance on entering its supported system LCO. When the loss of function is the result of multiple support systems, the appropriate LCO is the LCO for the supported system.

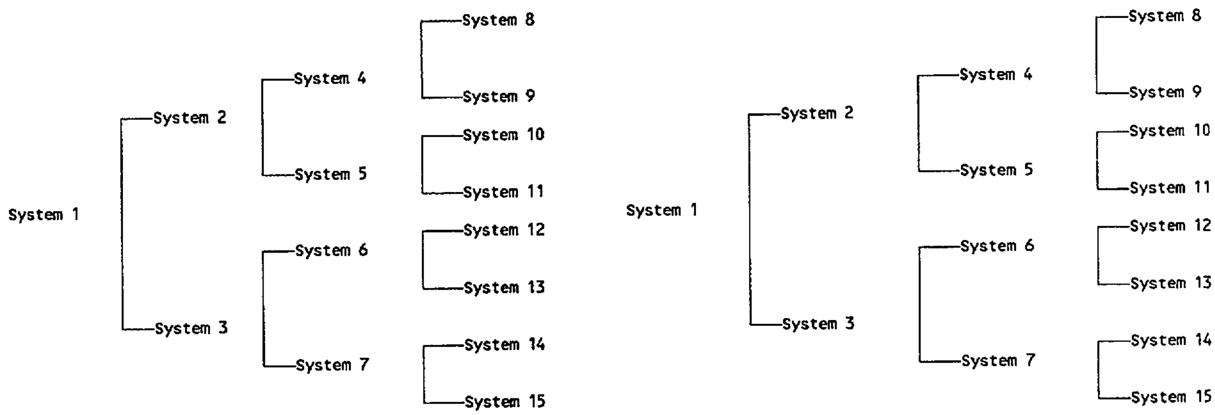
(continued)

BASES

LCO 3.0.6
(continued)

EXAMPLES

TRAIN A TRAIN B



EXAMPLE B3.0.6-1
If System 2 of Train A is inoperable, and System 5 of Train B is inoperable, a loss of safety function exists in supported Systems 5, 10 and 11.

EXAMPLE B3.0.6-2
If System 2 of Train A is inoperable, and System 11 of Train B is inoperable, a loss of safety function exists in System 11 which is in turn supported by System 5.

EXAMPLE B3.0.6-3
If System 2 of Train A is inoperable, and System 1 of Train B is inoperable, a loss of safety function exists in Systems 2,4,5,8,9,10, and 11.

For the examples above, support systems are to the left of the supported systems (i.e., System 1 supports System 2 and System 3).

(continued)

ENCLOSURE 3

PVNGS CENTS Topical
CEA Ejection Benchmark

1.1.1 CEA Ejection

1.1.1.1 Event Description

A CEA ejection event is a postulated limiting fault, which may result from a circumferential rupture of a CEDM housing or an associated nozzle. A CEA ejection will result in a rapid positive reactivity addition and power excursion that is initially mitigated by Doppler feedback and delayed neutron effects. A reactor trip on CPC VOPT is typically expected in a few seconds. Restricting CEA reactivity insertion, which limits the amount of reactivity that can be added upon ejection, minimizes fuel damage.

For the postulated CEA ejection event, it is assumed that the rupture that initiates the event is plugged by the ejected CEA, thereby allowing RCS pressure to rapidly increase until the PSVs open. Crediting HPPT for these cases would maximize the RCS peak pressure, since this trip comes in much after the VOPT. If it is assumed that the rupture is not plugged by the ejected CEA, then the NSSS response is bounded by SBLOCA analyses.

1.1.1.2 Methodology

The PVNGS NSSS response to a CEA ejection was simulated with the CENTS and CESEC computer codes. These benchmark analyses, which assessed the potential for RCS overpressurization, are summarized herein.

The CEA ejection cases presented in the PVNGS AOR simulates the event for fuel performance (fuel temperature and enthalpy) and RCS peak pressure evaluation. The fuel performance cases used the STRIKIN-II code, as required by Reference 1.1.1.7, whereas the RCS peak pressure case used CESEC. Therefore, only the RCS peak pressure case was selected for CENTS benchmarking purposes. The AOR was reanalyzed with both CENTS and CESEC. This reanalysis utilized the same initial and transient parameters and conditions that were used in previous bounding CEA ejection analyses. The fuel performance cases will continue to be analyzed using STRIKIN-II code as previously.

The ejected CEA causes a power excursion that rapidly adds energy to the RCS and, ultimately, the secondary system. The maximum RCS pressure must be less than the applicable ASME Section III limits.

1.1.1.3 Assumptions and Initial Conditions

The initial conditions from which the CENTS and CESEC codes were initiated are summarized in Table 1.1.1-1.

1.1.1.4 Benchmark Calculations

Table 1.1.1-2 presents a comparative sequence of events for the RCS peak pressure CENTS and CESEC benchmark analyses. Figures 1.1.1-1 through 1.1.1-9 show the comparative behavior of key parameters of interest, corresponding to the AOR reanalyses that are summarized in this table. The Figure 1.1.1-8 shows an increase in SG liquid mass for CESEC while a decrease in SG liquid mass of CENTS. This difference in SG liquid mass response is attributed to the difference in modeling of the two computer codes for the feedwater and steam flow post trip, CENTS code having more detailed modeling of the secondary system. As indicated by Figure 1.1.1-9, there is a difference in steam flow following the closure of the TAVs. Whereas CESEC models a sudden stop of steam flow, the CENTS code utilizes a more realistic model to reach pressure equilibrium between the SGs and the main steam headers. Therefore, another CENTS analysis was performed in which steam flow was quickly ramped down to zero following the trip, in an effort to better match the CESEC model. The comparative sequence of events for this RCS peak pressure benchmark is presented in Table 1.1.1-3, and the comparative behavior of key parameters of interest are illustrated in Figures 1.1.1-10 through 1.1.1-18.

1.1.1.5 Discussion of Results

It should be noted that different MSSV models are used in the CENTS and CESEC codes. Whereas MSSV flow areas and opening setpoints must be manually adjusted in CESEC to account for the pressure drop in the main steam headers, the CENTS code calculates this pressure drop automatically. This results in lower opening and closing pressures for the MSSVs, which are reflected in SG pressure as seen in Figures 1.1.1-6 and 1.1.1-15. It is noted from Figures 1.1.1-5 and 1.1.1-14 that there is a discrepancy in reactivities between CESEC and CENTS. The CESEC run undercredited the intended 5.5%$\Delta\rho$ scram reactivity due to error in the basedeck. Since this 'undercredit' occurs after the RCS peak pressure is reached, it only affects long term pressure response.

The second CENTS rerun with a steam ramp down following trip results in an increase of RCS peak pressure by 7.5 psi (see Tables 1.1.1-2 and 1.1.1-3).

1.1.1.6 Conclusions

The CESEC and CENTS benchmark analyses show generally good agreement in key parameters for the CEA ejection event. Both sets of benchmark analyses reveal that RCS peak pressure results are more benign when this event is modeled with the CENTS code. This is attributed primarily to more detailed modeling of the secondary system in CENTS, including the SGs and the main steam headers. Better heat removal in CENTS reduces RCS heatup and overpressurization, thereby resulting in a shorter delay in the HPPT and lower core inlet and outlet temperatures.

1.1.1.7 Reference

1. CENPD-190-A. "CE Method for Control Element Assembly Ejection Analysis." July 1976.

LIST OF ABBREVIATIONS

AOR	Analysis of Record
CEA	Control Element Assembly
CEDM	Control Element Drive Mechanism
CENTS	Combustion Engineering Nuclear Transient Simulation
CESEC	Combustion Engineering NSSS Simulation Code
CPC	Core Protection Calculator
HPPT	High Pressurizer Pressure Trip
MSSV	Main Steam Safety Valve
MTC	Moderator Temperature Coefficient
NSSS	Nuclear Steam Supply System
PSV	Pressurizer Safety Valve
PVNGS	Palo Verde Nuclear Generating Station
RCS	Reactor Coolant System
SBLOCA	Small Break LOCA
SG	Steam Generator
STRIKIN-II	Hot Rod Heatup Computer Code
TAV	Turbine Admission Valve
VOPT	Variable Over Power Trip

Table 1.1.1-1
Assumptions and Initial Conditions for the CEA Ejection RCS Peak Pressure CENTS and CESEC Benchmark Analyses

Parameter	Value
Core Power (MWt)	3980
Core Inlet Temp (^oF)	548
Initial Pressurizer Pressure (psia)	2100
Initial RCS Flow (lbm/sec)	43278
Initial Pressurizer Level (ft)	15.2/15.1
Initial SG Liquid Mass (lbm)	155,982/155,300
MTC ($x10^{-4}$ $\%\Delta\rho/^oF$)	0.0
H-gap (Btu/hr-ft^2-^oF)	6984
Plugged SG Tubes	1500
ASI	+0.3
Ejection Time (sec)	0.05
Ejected Rod Worth at full power ($\%\Delta\rho$)	0.157
CEA Worth for Trip ($\%\Delta\rho$)	-5.5

Table 1.1.1-2
Sequence of Events for the CEA Ejection RCS Peak Pressure CENTS and CESEC Benchmark Analyses

Time (sec)		Event	Setpoint or Value	
CESEC	CENTS		CESEC	CENTS
0.00	0.00	Mechanical Failure of CEDM Causes CEA to Eject	---	---
0.05	0.05	CEA Fully Ejected	---	---
0.18	0.07	Maximum Core Power (% of Design Power)	151.2	149.5
11.70	15.28	Pressurizer Pressure Reaches Trip Setpoint (psia)	2450	2450
12.45	16.03	HPPT, Turbine Trip, and MFW Trip Occurs	---	---
14.05	17.60	PSVs Open (psia)	2550	2550
14.45	18.00	Maximum RCS Pressure (psia)	2656.2	2641.1
15.05	19.80	PSVs Closed (psia)	2486.3	2486.3
38.30	23.80	Bank 1 MSSVs Open (psia)	1303.0	1303.0
---	---	Bank 2 MSSVs Open (psia)	1344.0	1344.0
---	---	Bank 3 MSSVs Open (psia)	1391.0	1391.0

Table 1.1.1-3
Sequence of Events for the
CEA Ejection RCS Peak Pressure
CENTS and CESEC Benchmark Analyses
(With CENTS Steam Flow Ramp Down Following Trip)

Time (sec)			Setpoint or Value	
CESEC	CENTS	Event	CESEC	CENTS
0.00	0.00	Mechanical Failure of CEDM Causes CEA to Eject	---	---
0.05	0.05	CEA Fully Ejected	---	---
0.18	0.07	Maximum Core Power (% of Design Power)	151.2	149.5
11.70	15.28	Pressurizer Pressure Reaches Trip Setpoint (psia)	2450	2450
12.45	16.03	HPPT, Turbine Trip, and MFW Trip Occurs	---	---
14.05	17.50	PSVs Open (psia)	2550	2550
14.45	17.94	Maximum RCS Pressure (psia)	2656.2	2648.6
15.05	19.83	PSVs Closed (psia)	2486.3	2486.3
38.30	22.64	Bank 1 MSSVs Open (psia)	1303.0	1303.0
---	---	Bank 2 MSSVs Open (psia)	1344.0	1344.0
---	---	Bank 3 MSSVs Open (psia)	1391.0	1391.0

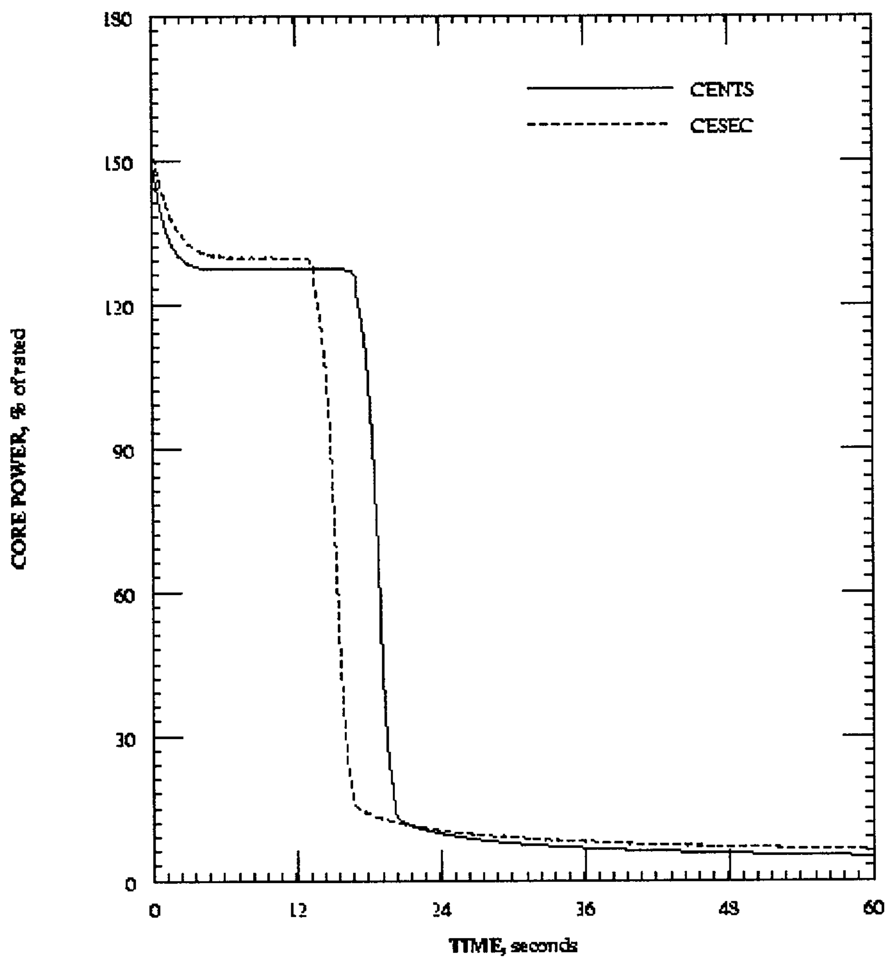


Figure 1.1.1-1
Core Power vs. Time
CEA Ejection RCS Peak Pressure Analyses

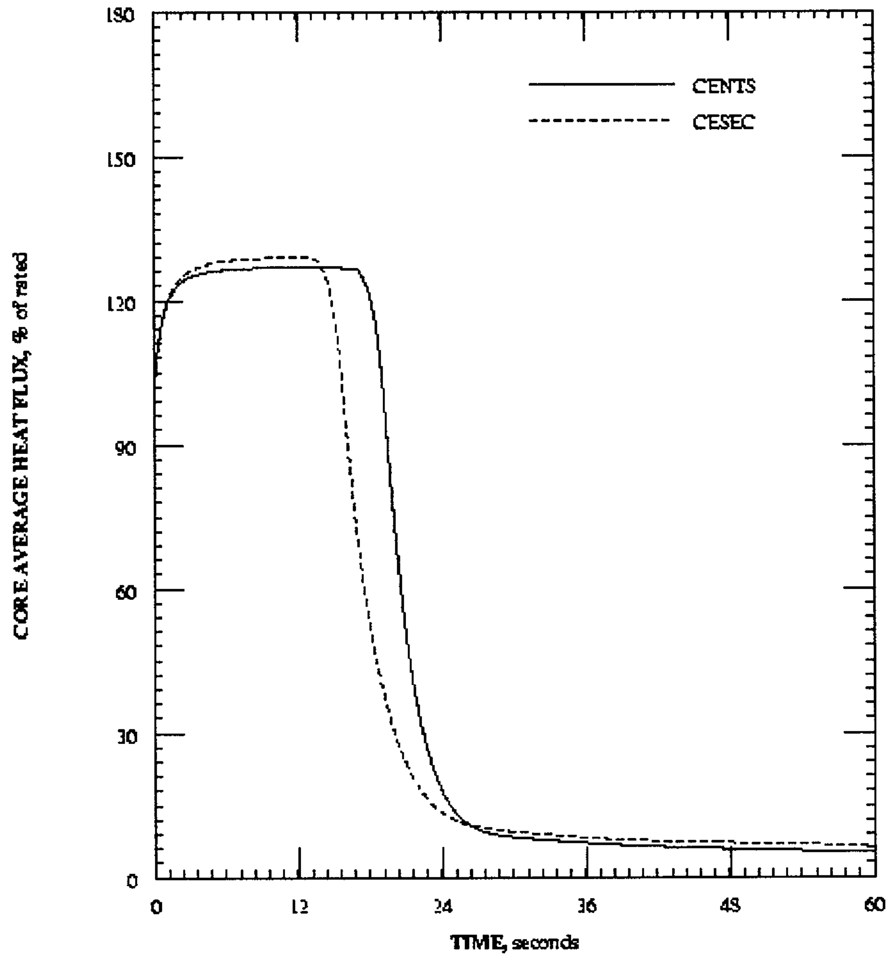


Figure 1.1.1-2
Core Average Heat Flux vs. Time
CEA Ejection RCS Peak Pressure Analyses

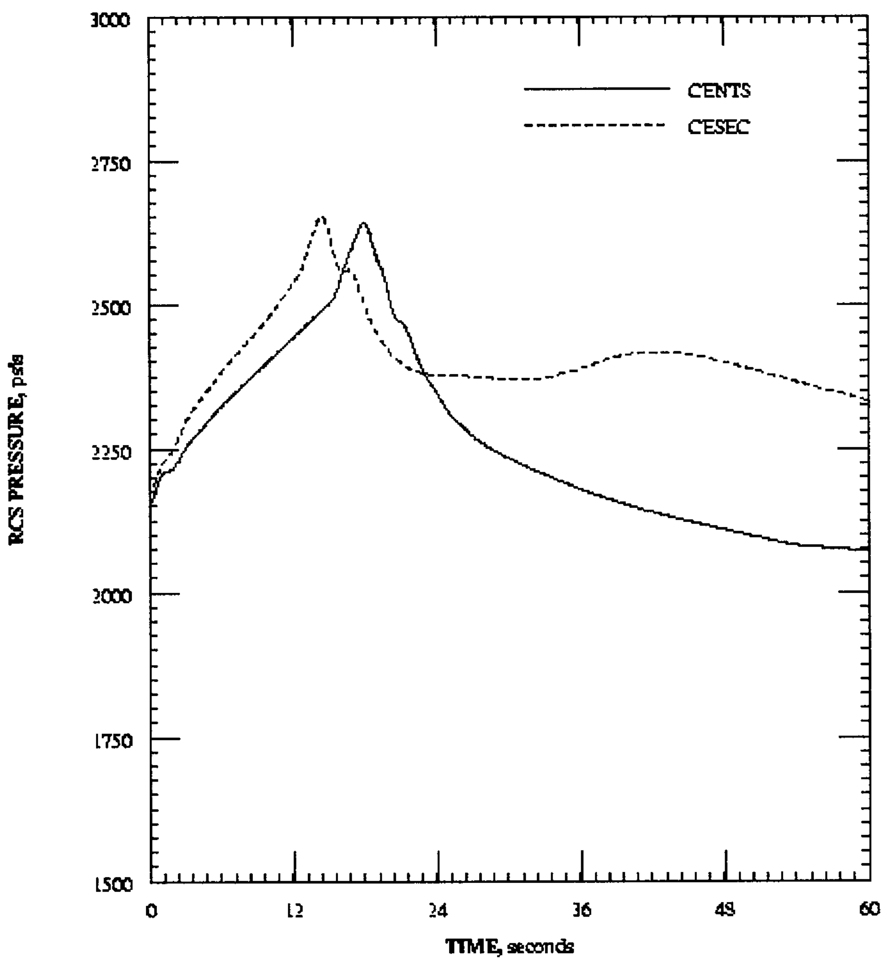


Figure 1.1.1-3
RCS Pressure vs. Time
CEA Ejection RCS Peak Pressure Analyses

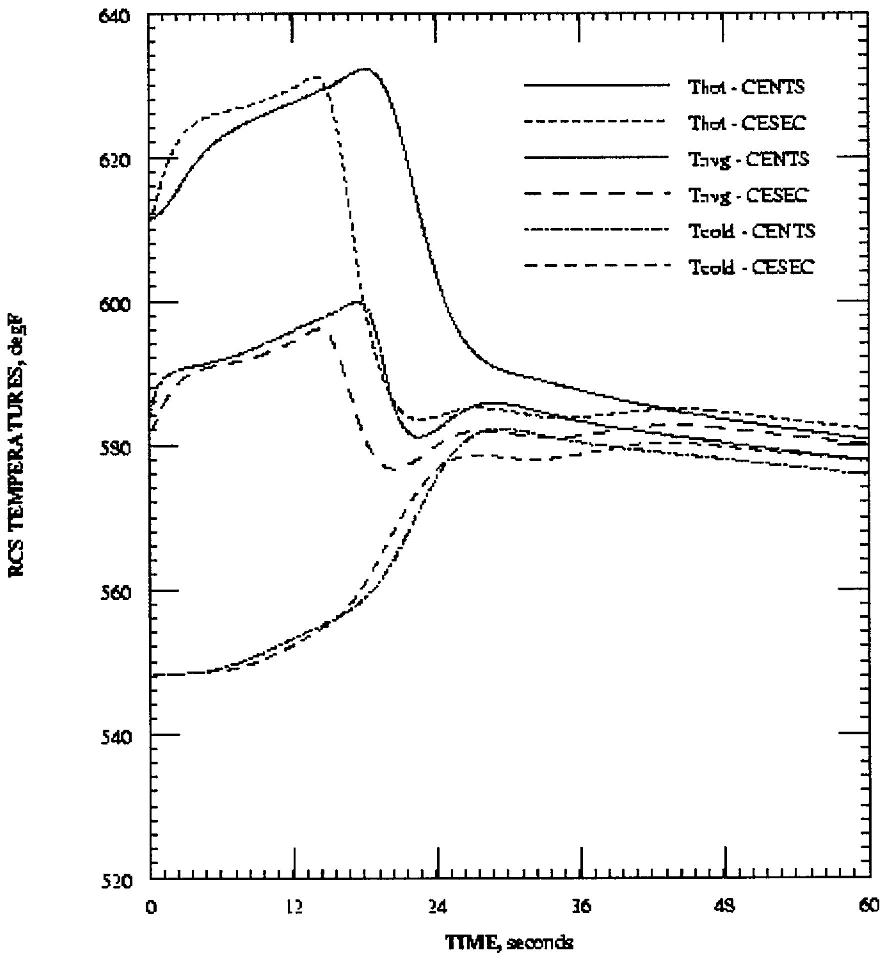


Figure 1.1.1-4
RCS Temperature vs. Time
CEA Ejection RCS Peak Pressure Analyses

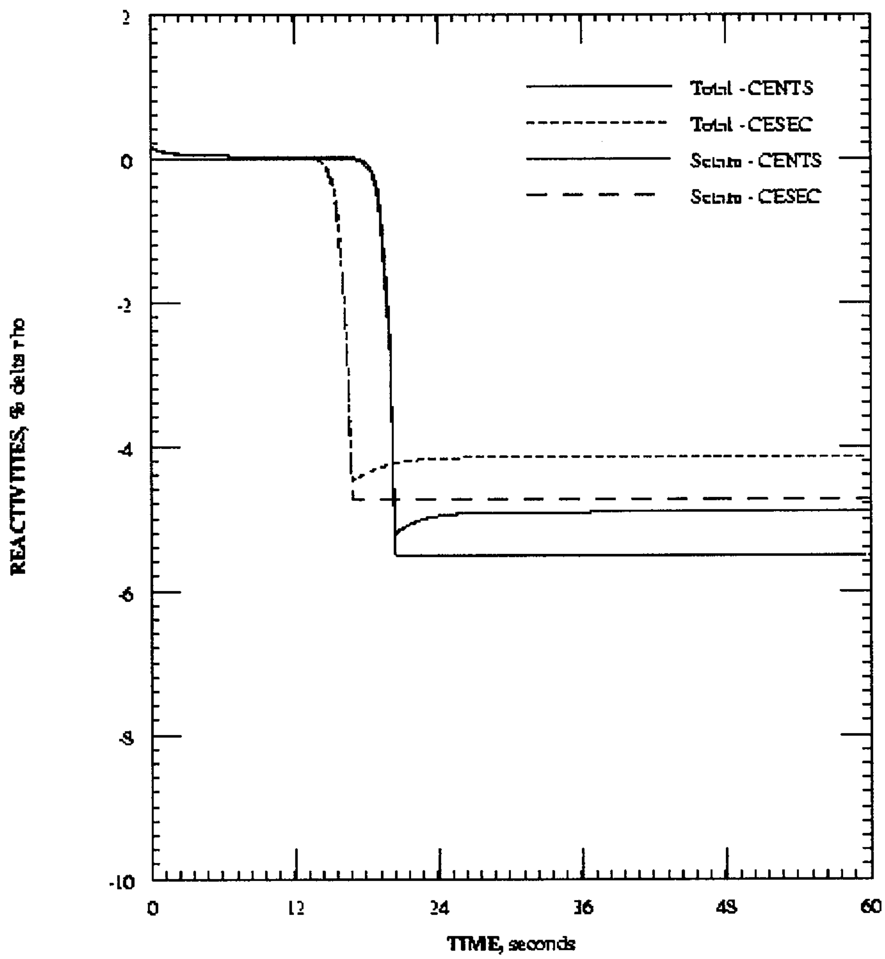


Figure 1.1.1-5
Reactivities vs. Time
CEA Ejection RCS Peak Pressure Analyses

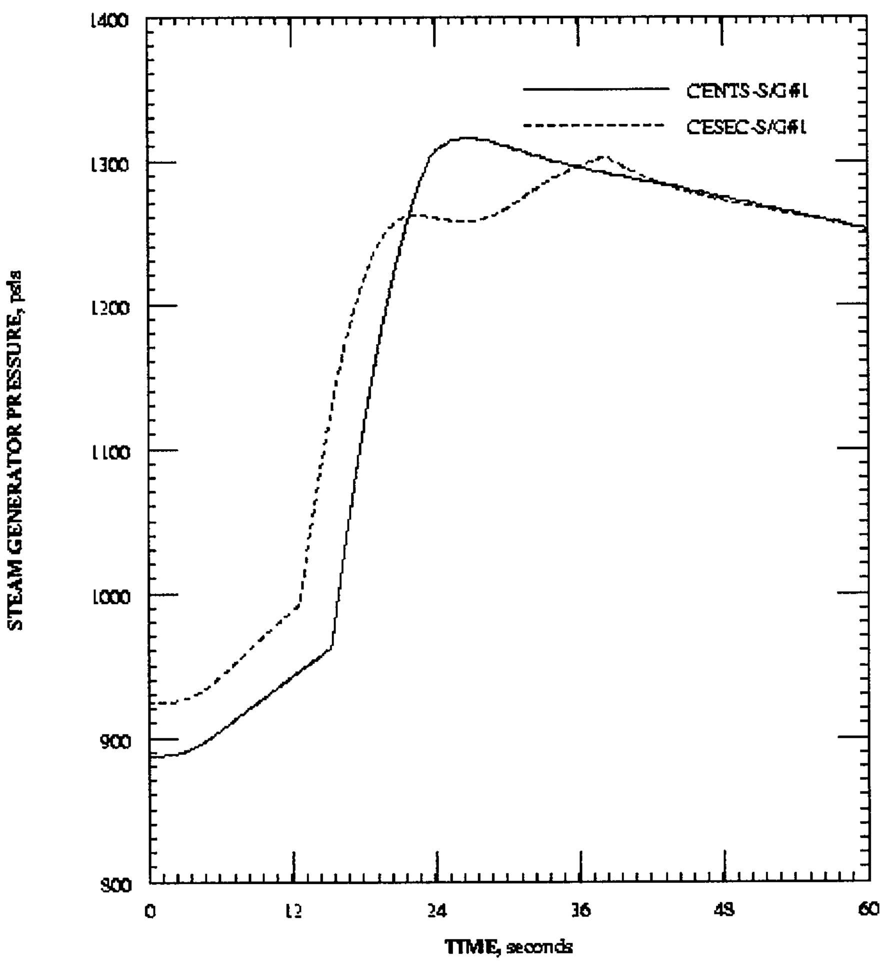


Figure 1.1.1-6
SG Pressure vs. Time
CEA Ejection RCS Peak Pressure Analyses

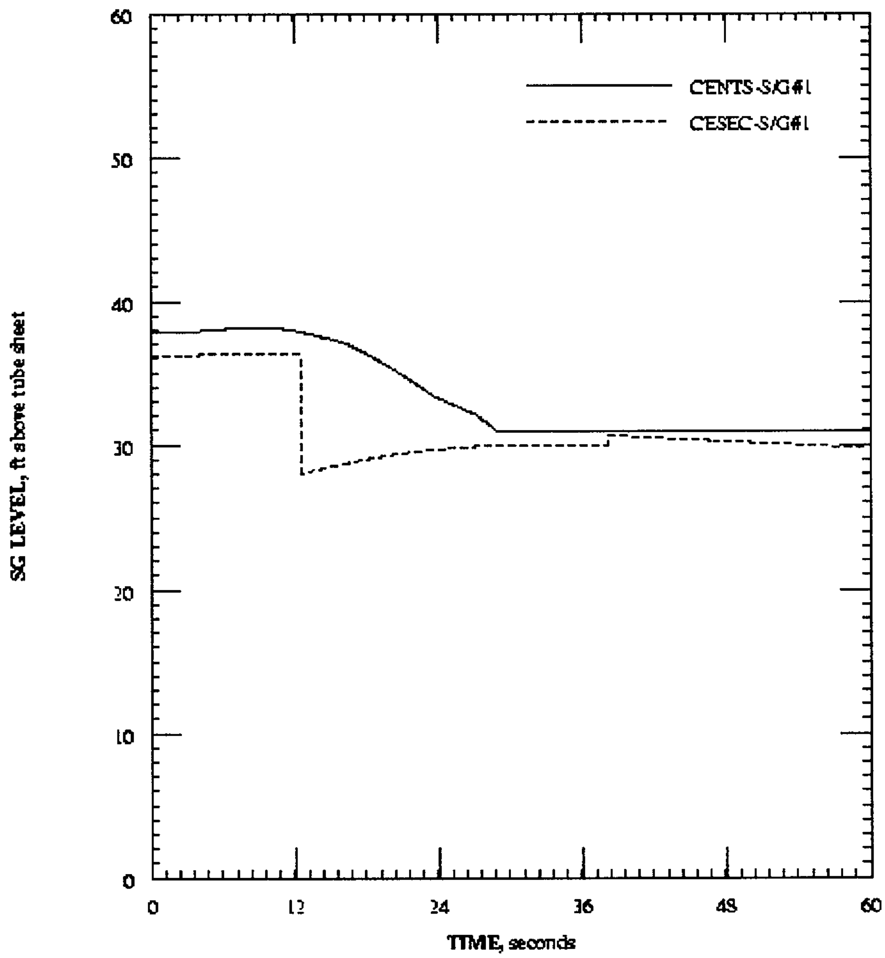


Figure 1.1.1-7
SG Level vs. Time
CEA Ejection RCS Peak Pressure Analyses

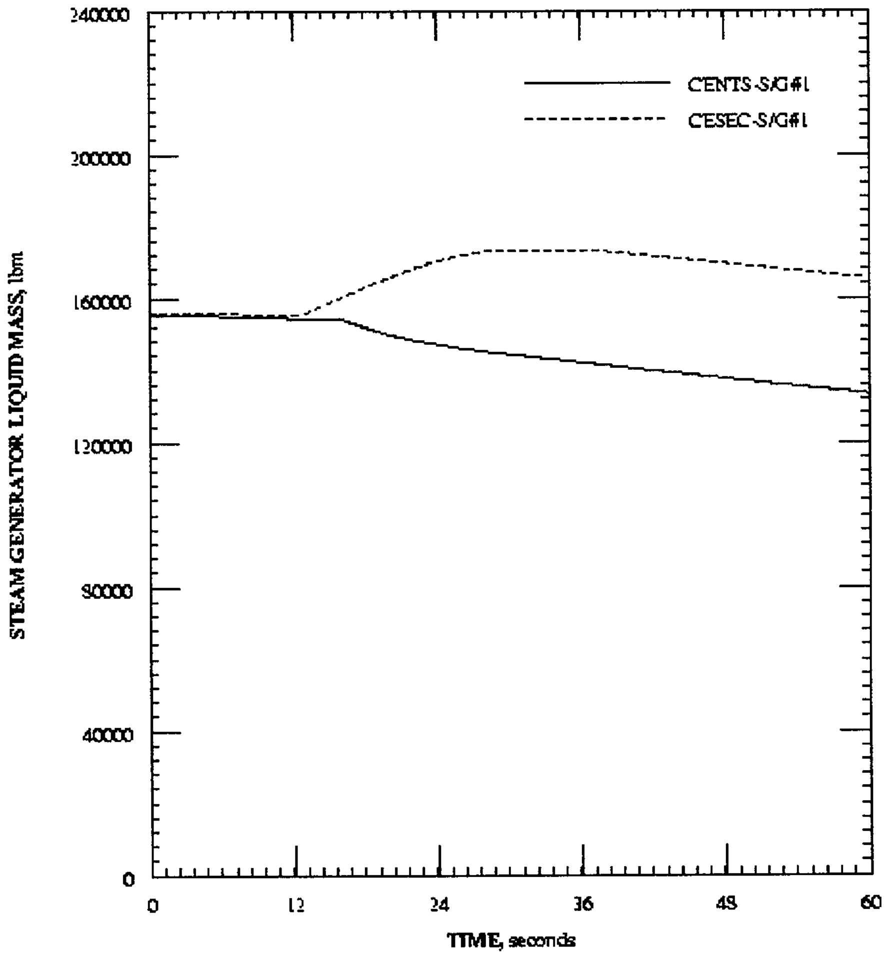


Figure 1.1.1-8
SG Mass vs. Time
CEA Ejection RCS Peak Pressure Analyses

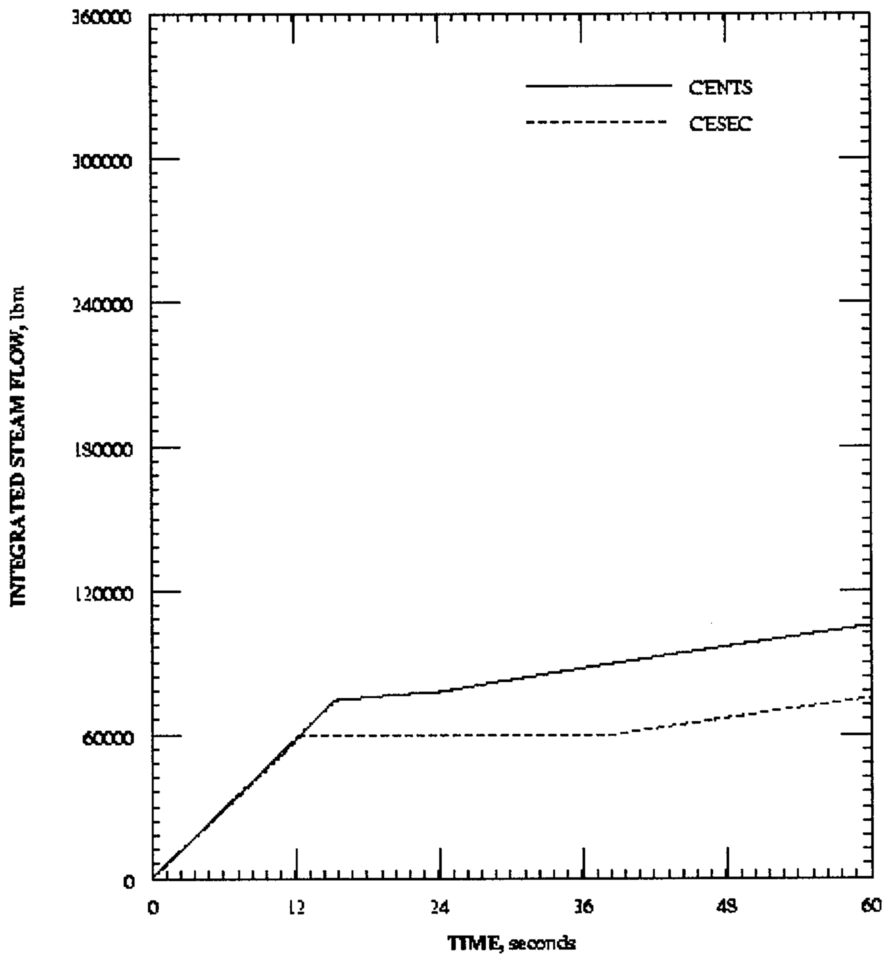


Figure 1.1.1-9
Integrated Steam Flow vs. Time
CEA Ejection RCS Peak Pressure Analyses

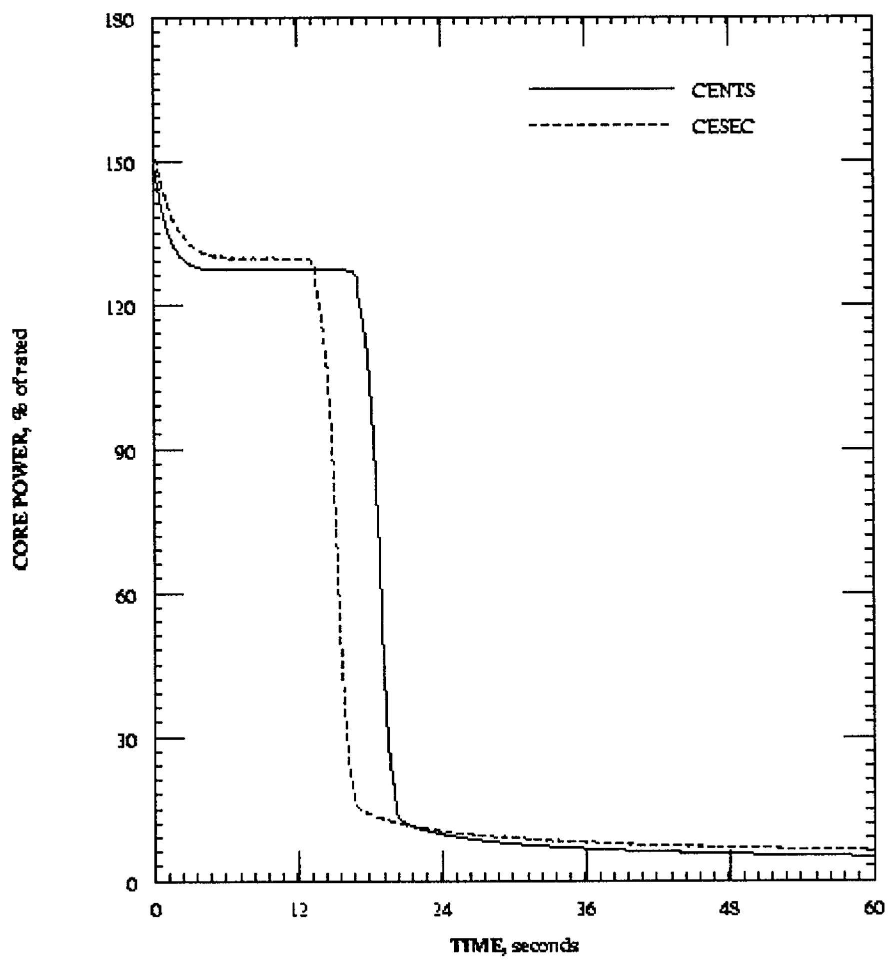


Figure 1.1.1-10
Core Power vs. Time
CEA Ejection RCS Peak Pressure Analyses
(With CENTS Steam Flow Ramp Down Following Trip)

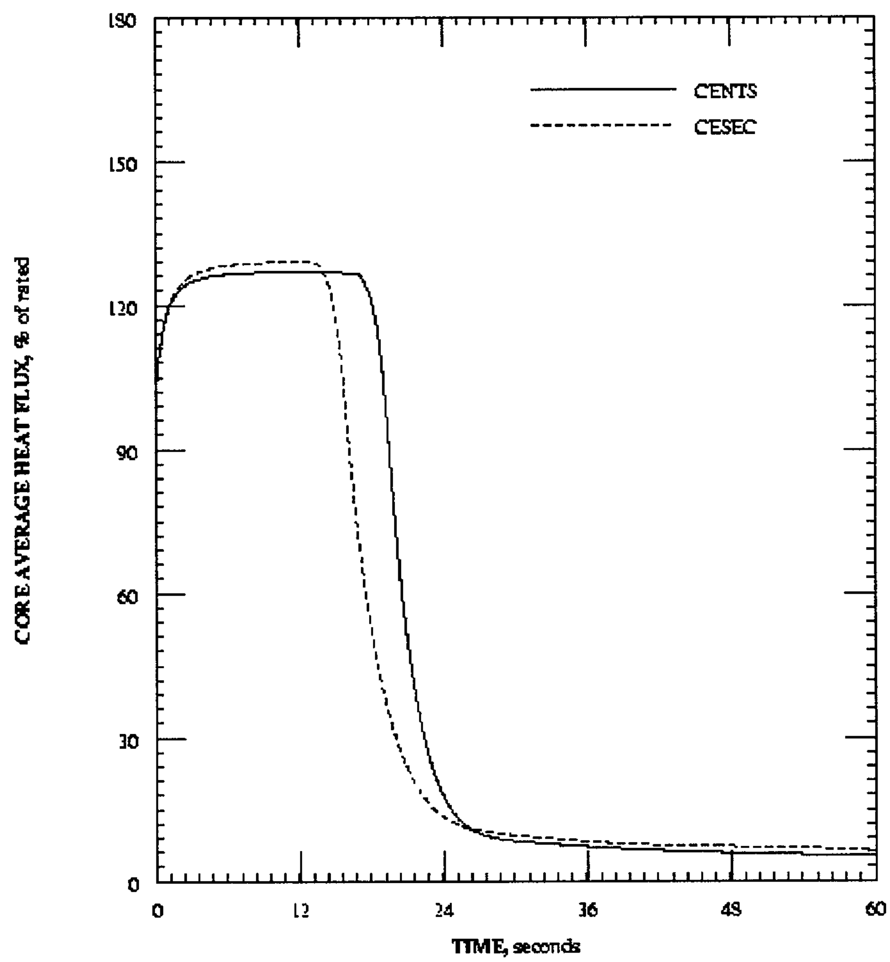


Figure 1.1.1-11
Core Average Heat Flux vs. Time
CEA Ejection RCS Peak Pressure Analyses
(With CENTS Steam Flow Ramp Down Following Trip)

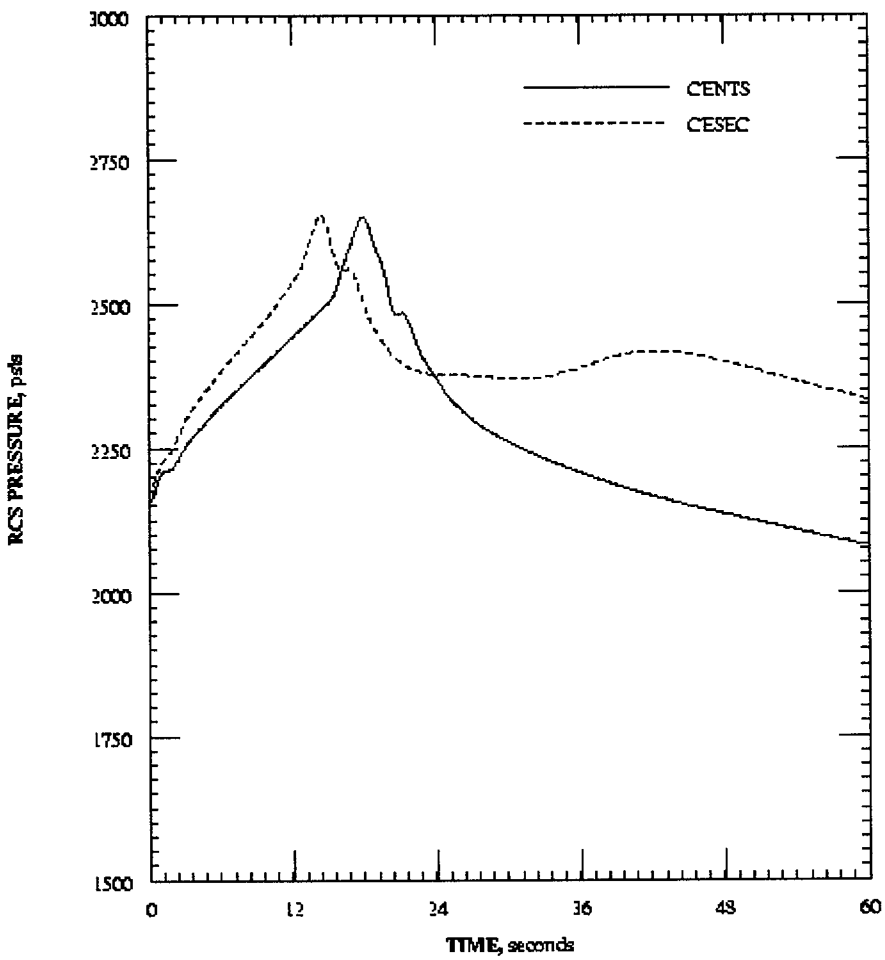


Figure 1.1.1-12
RCS Pressure vs. Time
CEA Ejection RCS Peak Pressure Analyses
(With CENTS Steam Flow Ramp Down Following Trip)

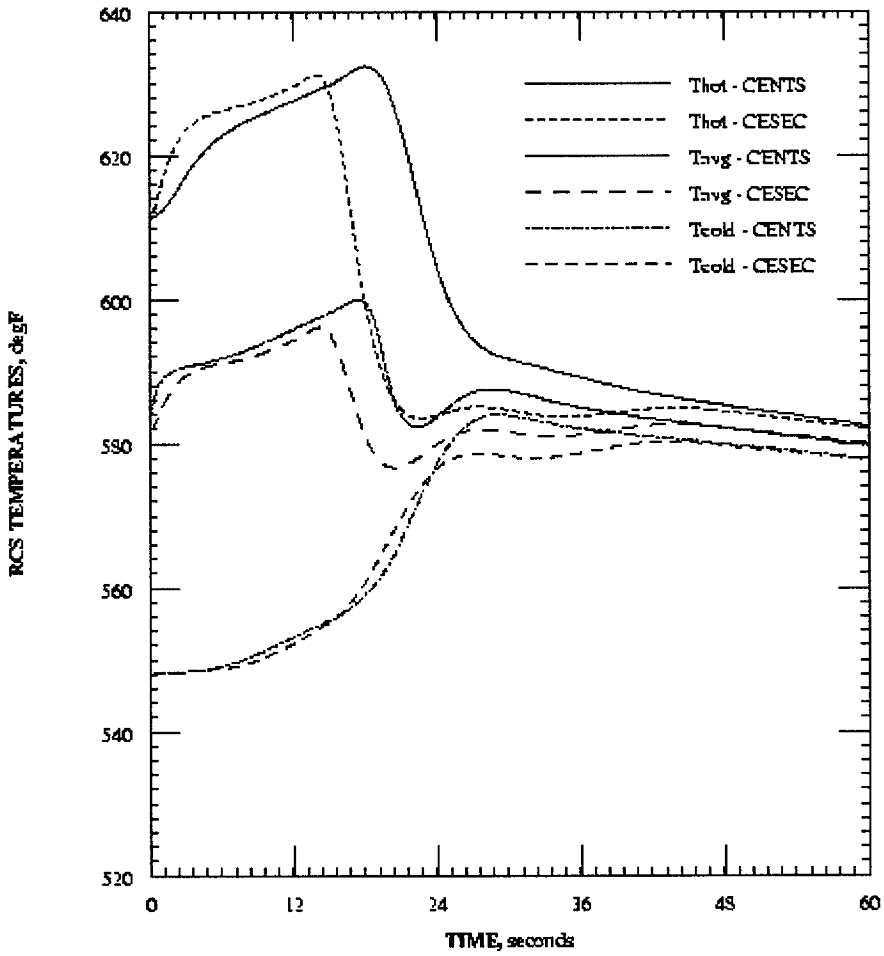


Figure 1.1.1-13
RCS Temperature vs. Time
CEA Ejection RCS Peak Pressure Analyses
(With CENTS Steam Flow Ramp Down Following Trip)

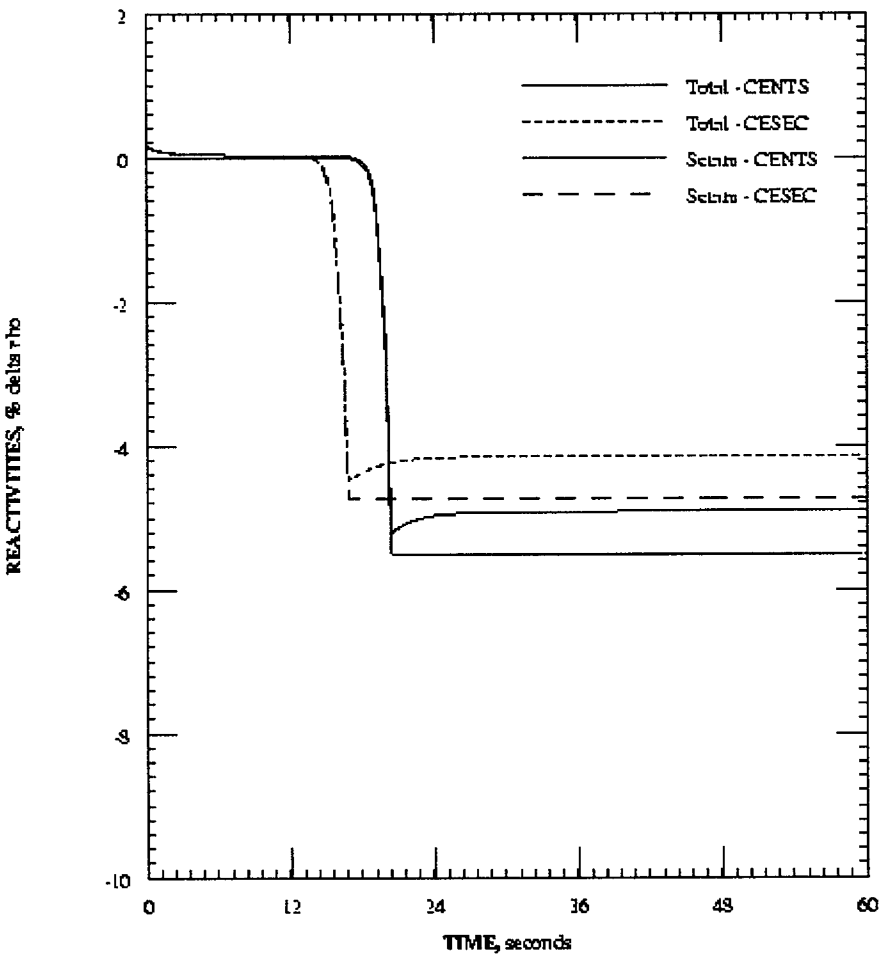


Figure 1.1.1-14
Reactivities vs. Time
CEA Ejection RCS Peak Pressure Analyses
(With CENTS Steam Flow Ramp Down Following Trip)

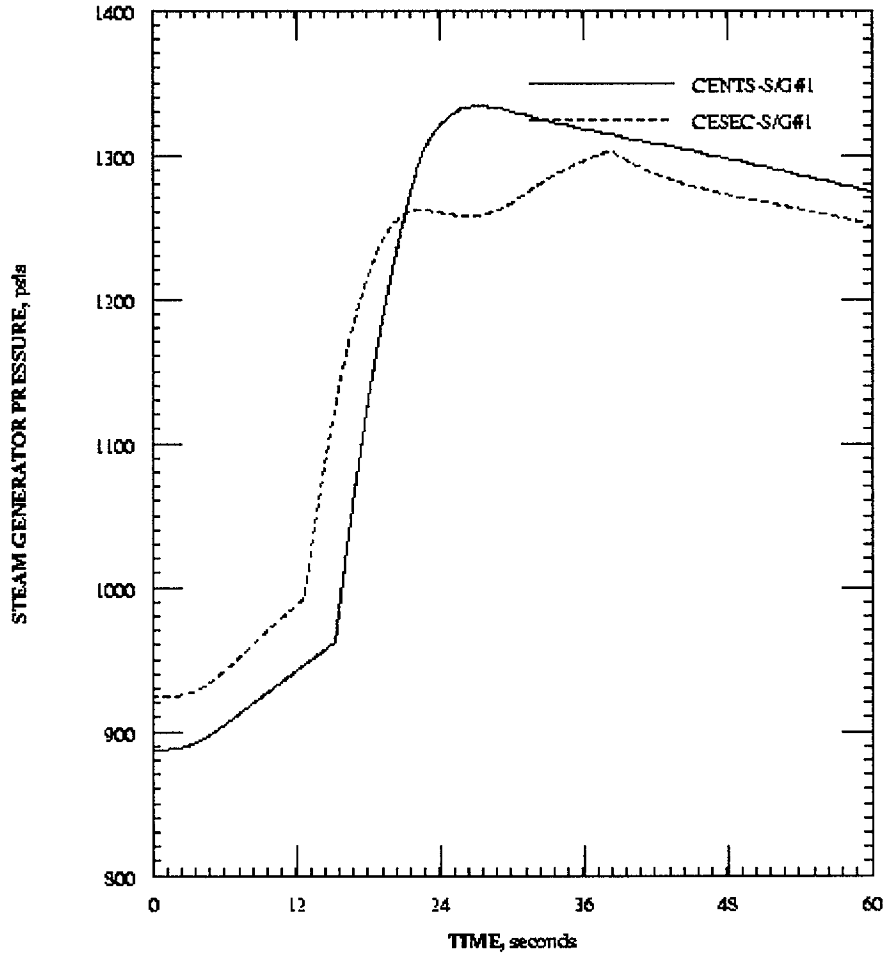


Figure 1.1.1-15
SG Pressure vs. Time
CEA Ejection RCS Peak Pressure Analyses
(With CENTS Steam Flow Ramp Down Following Trip)

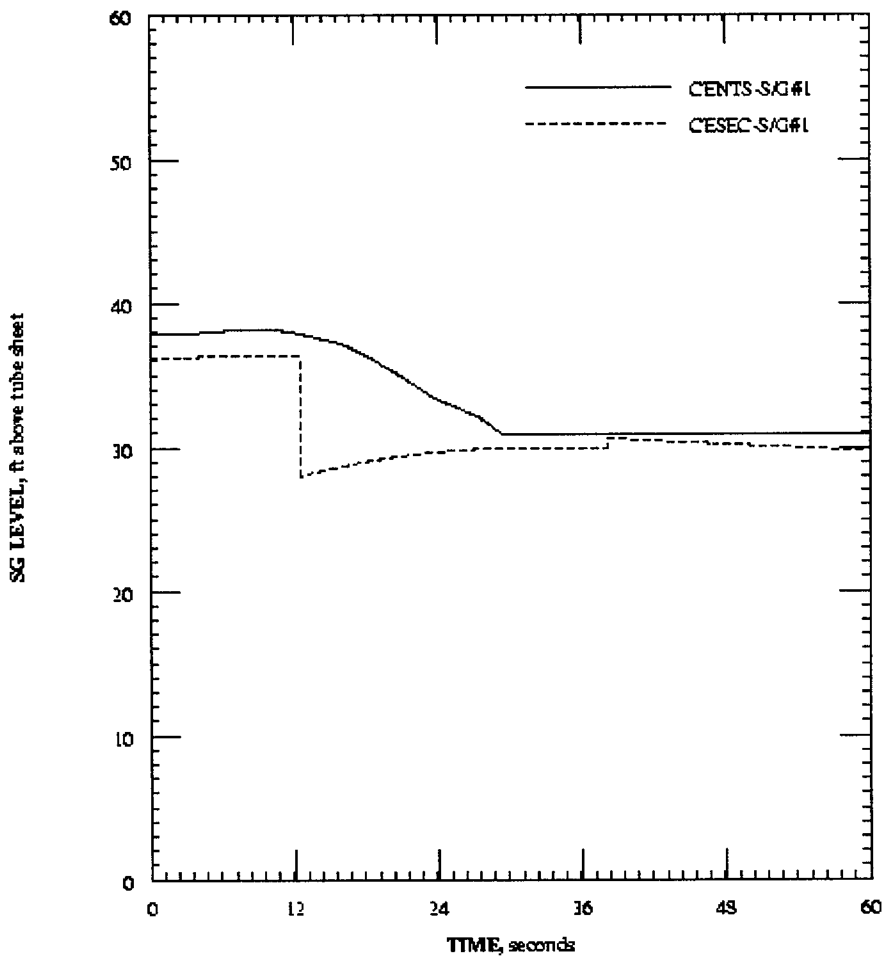


Figure 1.1.1-16
SG Level vs. Time
CEA Ejection RCS Peak Pressure Analyses
(With CENTS Steam Flow Ramp Down Following Trip)

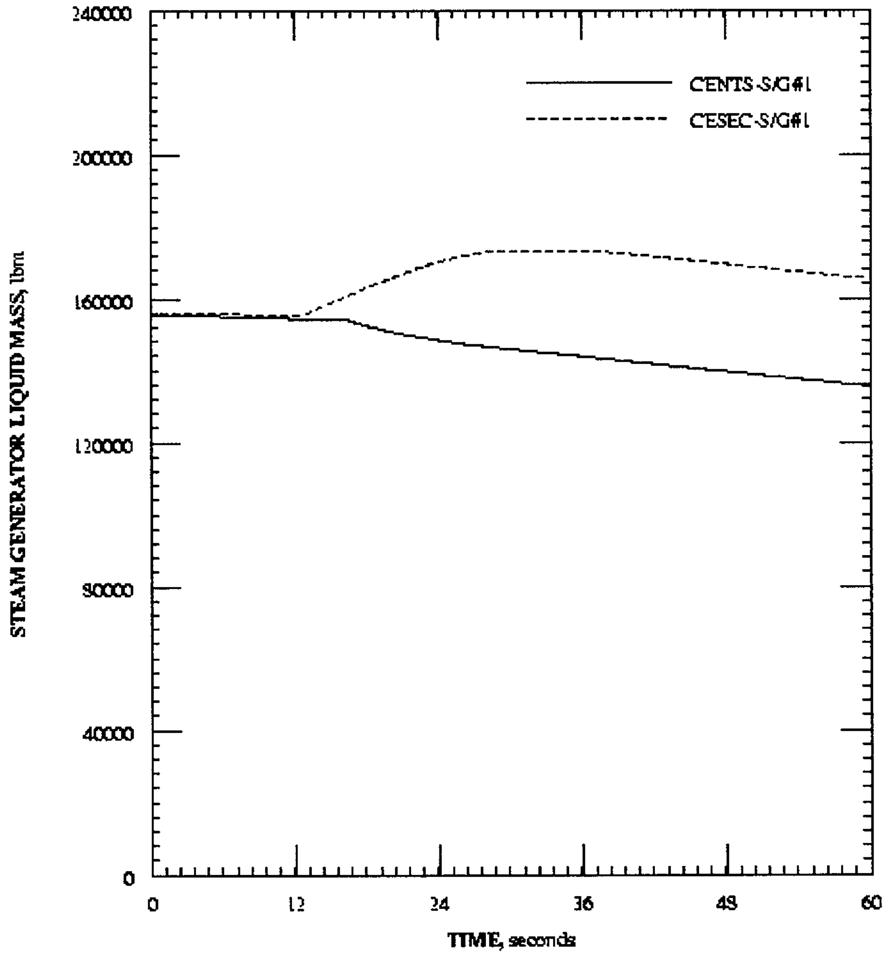


Figure 1.1.1-17
SG Mass vs. Time
CEA Ejection RCS Peak Pressure Analyses
(With CENTS Steam Flow Ramp Down Following Trip)

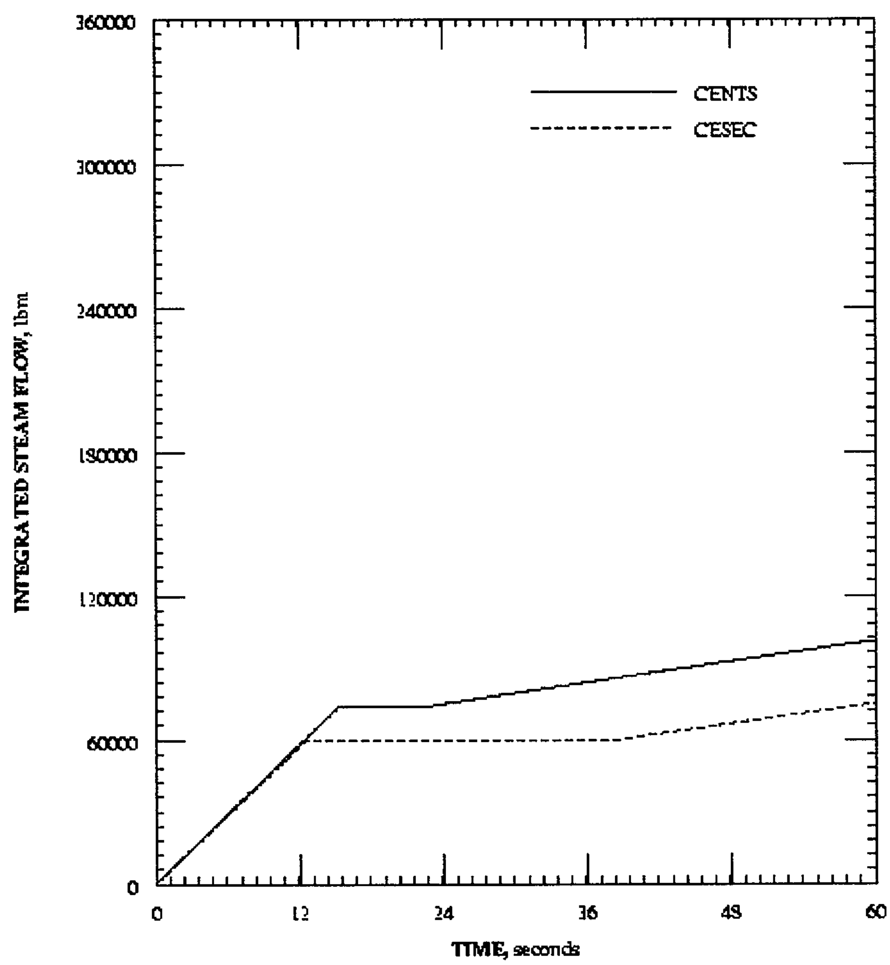


Figure 1.1.1-18
Integrated Steam Flow vs. Time
CEA Ejection RCS Peak Pressure Analyses
(With CENTS Steam Flow Ramp Down Following Trip)

ENCLOSURE 4

Full Text Topical Report References from Technical Specification 5.6.5.b to be located in COLR
(for information only)

COLR References

TS 5.6.5.b

1. "CE Method for Control Element Assembly Ejection Analysis, "CENPD-0190-A, January 1976 (Methodology for Specification 3.1.7, Regulating CEA Insertion Limits).

2. "The ROCS and DIT Computer Codes for Nuclear Design," CENPD-266-P-A, April 1983 [Methodology for Specifications 3.1.1, Shutdown Margin - Reactor Trip Breakers Open; 3.1.2, Shutdown Margin - Reactor Trip Breakers Closed; 3.1.4, Moderator Temperature Coefficient BOL and EOL limits; 3.1.7, Regulating CEA Insertion Limits and 3.9.1, Boron Concentration (Mode 6)].

4. "Modified Statistical Combination of Uncertainties," CEN-356(V)-P-A Revision 01-P-A, May 1988 and "System 80™ Inlet Flow Distribution," Supplement 1-P to Enclosure 1-P to LD-82-054, February 1993 (Methodology for Specification 3.2.4, DNBR and 3.2.5 Axial Shape Index).

5. "Calculative Methods for the CE Large Break LOCA Evaluation Model for the Analysis of CE and W Designed NSSS," CENPD-132, Supplement 3-P-A, June 1985 (Methodology for Specification 3.2.1, Linear Heat Rate).

6. "Calculative Methods for the CE Small Break LOCA Evaluation Model," CENPD-137-P, Supplement 1P, January 1977 (Methodology for Specification 3.2.1, Linear Heat Rate)

9. "Fuel Rod Maximum Allowable Pressure," CEN-372-P-A, May 1990 (Methodology for Specification 3.2.1, Linear Heat Rate).

11. "Arizona Public Service Company PWR Reactor Physics Methodology Using CASMO-4/SIMULATE-3," September 1999 [Methodology for Specifications 3.1.1, Shutdown Margin - Reactor Trip Breakers Open; 3.1.2, Shutdown Margin - Reactor Trip Breakers Closed; 3.1.4, Moderator Temperature Coefficient; 3.1.7, Regulating CEA Insertion Limits and 3.9.1, Boron Concentration (Mode 6)].

12. "Technical Manual for the CENTS Code," CE-NPD 282-P-A, Volumes 1-3, October 1991 [Methodology for Specifications 3.1.2, Shutdown Margin-Reactor Trip Breakers Closed; 3.1.4, Moderator Temperature Coefficient; 3.1.5, CEA Alignment; 3.1.7, Regulating CEA Insertion Limits; 3.1.8, Part Length CEA Insertion Limits and 3.2.3, Azimuthal Power Tilt- T_q].