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FACIL: STN-50-528 Palo Verde Nuclear Station, Unit 1, Arizona Publi 05000528
STN-50-529 Palo Verde Nuclear Station, Unit 2, Arizona Publi 05000529
STN-50-530 Palo Verde Nuclear Station, Unit 3, Arizona Publi 05000530
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RECIP. NAME RECIPIENT AFFILIATION
Document Control Branch (Document Control Desk)

SUBJECT: Forwards addl info re 870123 application for amends to Licenses NPF-41, NPF-51 & NPF-65, reducing boration requirements when in shutdoun, per 870303 telcon & 870401 ltr.

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April 23, 1987
161-00167-JGH/PGN

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington D.C. 20555

Subject: Palo Verde Nuclear Generating Station (PVNGS)
Units 1, 2, and 3
Docket Nos. STN 50-528 (License NPF-41)
STN 50-529 (License NPF-51)
STN 50-530 (License NPF-65)
Revision to Technical Specification Amendment
Sections 1.0, 2.2, 3/4.1, 3/4.3, 3/4.10
File: 87-F-005-419.05; 87-A-056-026

Reference: (1) Letter from J. G. Haynes (ANPP) to G. W. Knighton (NRC) dated January 23, 1987 (PP39798). Subject: Technical Specification Amendment - Sections 1.0, 2.2, 3/4.1, 3/4.3, 3/4.10.

(2) Telecon between Manny Licitra (NRC), Peggy Nelson (ANPP), Jim Webb (ANPP), and Dave Hoppes (ANPP) on March 3, 1987. Subject: Temperature Dependent Shutdown Margin Technical Specification Amendment Request.

(3) Letter from E. A. Licitra (NRC) to E. E. Van Brunt, Jr. (ANPP) dated April 1, 1987. Subject: Request for Additional Information - Palo Verde Units 1, 2 and 3 Shutdown Margin Technical Specification Change Request.

Dear Sir:

The Reference (1) letter submitted proposed changes to the PVNGS Units 1 and 2 Technical Specifications to reduce the boration requirements when in shutdown, by modification of the shutdown margin requirements. In addition, it was requested that the change be considered as a change to the PVNGS Unit 3 Technical Specifications.

The Reference (2) telecon and Reference (3) letter requested additional information on the amendment request. This information is provided in Attachment 1.

Attachment 2 provides revised pages to the Reference (1) letter, reflecting the additional information discussed in Attachment 1. The revised pages replace the corresponding pages in the original submittal.

By copy of this letter, we are also forwarding the revisions to the appropriate state agency.

A001
1/1

If you have any questions, please call W. F. Quinn at (602) 371-4087.

Very truly yours,

J. G. Haynes
Vice President
Nuclear Production

JGH/PGN/ls
Attachment

cc: O. M. De Michele (all w/a)
E. E. Van Brunt, Jr.
A. C. Gehr
R. P. Zimmerman
J. B. Martin
G. W. Knighton
C. E. Tedford

ATTACHMENT 1
ANPP RESPONSE TO NRC REQUEST FOR ADDITIONAL INFORMATION

1. NRC QUESTION

Should not the new Definition 1.16 read "...assuming the fully or partially withdrawn full-length CEA..." rather than "...assuming the fully or partially inserted full-length CEA.."?

ANPP RESPONSE

The definition is correct as submitted since the proposed wording would lead to the phrase "...withdrawn full-length CEA ...is withdrawn."

2. NRC QUESTION

Many of the Figures accompanying the supporting analyses for the Technical Specification amendment request are unlabeled and some are also completely blank (e.g., Fig. 1-1, Fig. 1-2, Fig. 1-15). Please correct and update these.

ANPP RESPONSE

Section H of the attachment to Reference (1) supplied the supporting analyses for the Technical Specification amendment request. Figures 1-1, 1-2, and 1-15 showed core power, core heat flux, and safety injection flow, respectively, as a function of time. These figures were for a large steam line break during Mode 3. As would be expected for Mode 3, core power and core heat flux are zero for the duration of the transient. In addition, the analysis states that during the first 500 seconds of the transient the pressurizer pressure remains above the Safety Injection Actuation Signal setpoint. Therefore, safety injection flow is also zero for the duration of the transient. These figures were originally scaled such that the curve was superimposed on the horizontal axis and it appeared that the curve was omitted from the figure. To facilitate the review of this amendment request, the figures have been rescaled such that the curves do not lie on the axis. Similarly, Figures 1-17, 1-18 and 1-31 have also been rescaled.

3. NRC QUESTION

For the proposed 1% shutdown margin when all CEAs are fully inserted, how is verification made that all CEAs are inserted? What instrumentation and/or read-out devices are used? Should this be included in Technical Specifications (e.g., 4.1.1.1.1.a.2)?

ANPP RESPONSE

Verification that all CEAs are fully inserted will be done via the rod bottom contacts, or if these are unavailable, via the reed switch position transmitters (RSPT). The rod bottom contacts are dedicated contacts, one for each CEA. The rod bottom contact for a CEA is closed when its associated CEA is fully inserted, and provides a readout of actual CEA position. The information is displayed to the operator in the control room via a core mimic

display in which an illuminated light indicates a CEA fully inserted. The rod bottom information is also available from the plant monitoring computer. If the rod bottom contacts are unavailable, then information from the RSPTs may be used as an alternate. There are two redundant, independent channels of RSPTs available. The information from these is available through the Control Element Assembly Calculator (CEAC) operator's modules in the control room. If the above noted indication has been made inoperable due to plant condition, e.g., during MODE 5 with RSPTs disconnected for maintenance, the Reactor Trip Breakers will be opened to insure the CEAs are fully inserted. This will be a part of the revised shutdown margin procedures which will implement the revised Technical Specification. Due to the diversity of CEA position indication which is available, it is not neccessary to include within the Technical Specification the method to be used in verifying that all CEAs are fully inserted.

4. NRC QUESTION

Provide justification for deleting any reference to shutdown margin requirements in the Action statements for Technical Specifications 3.1.2.2, 3.1.2.4, 3.1.2.6.

ANPP RESPONSE

The Action statements for Technical Specifications 3.1.2.2, 3.1.2.4, and 3.1.2.6 currently state ". . . .or be in at least HOT STANDBY and borated to a SHUTDOWN MARGIN equivalent to at least 6% $\Delta k/k$ at 210 F° within the next 6 hours . . . ." The reference to shutdown margin requirements in the action statements for Technical Specifications 3.1.2.2, 3.1.2.4, and 3.1.2.6 is redundant, since the requirement to be in HOT STANDBY automatically requires boration to a shutdown margin consistent with Technical Specification 3.1.1.1 or 3.1.1.2.

5. NRC QUESTION

Discuss how the limiting CEA ejection event was reevaluated concurrent with the introduction of the K_{N-1} requirement to ensure that the safety analysis acceptance criteria will be met.

ANPP RESPONSE

The K_{N-1} requirement was introduced to preclude additional evaluation of the CEA ejection event. If the K_{N-1} requirement is met, by definition, the reactor will remain subcritical if a single rod ejection occurs. Safety analysis acceptance criteria are met without further analysis.

5. NRC COMMENT

In the No Significant Hazards Consideration (NSHC) Determination, changes 3, 4, and 5 are not administrative changes, but are classified as such.

ANPP RESPONSE

In the NSHC Dermination, changes 3, 4, and 5 were determined to match the guidance of 51 FR 7751 by the example: (i) A purely administrative change to T.S. These changes involved:

3. Removing the redundant reference to the shutdown margin requirement in T.S. 3.1.2.2, 3.1.2.4 and 3.1.2.6.

4. Revising the reference to shutdown margin in T.S. 3.10.1 to be consistent with the new shutdown margin requirements.

5. Provide a special test exception to suspend shutdown margin requirements for pre-startup testing of the control element drive mechanism system.

These changes were determined to be purely administrative since they are only necessary to maintain consistency throughout the Technical Specifications with the addition of the new shutdown margin requirements. However, to facilitate the review of this amendment request, changes 3, 4 and 5 will be classified as both an administrative change (example (i)) and: A change which may result in some increase to the probability or consequences of a previously analyzed accident, or may reduce in some way a safety margin, but where the results of the change are clearly within all acceptable criteria with respect to the system or component specified in the Standard Review Plan (example (iv)).

Accordingly, the NSHC Determination has been revised to reflect this change.

ATTACHMENT 2
Revisions to Temperature Dependent Shutdown
Margin Technical Specification Amendment Request

ATTACHMENT

A. DESCRIPTION OF THE TECHNICAL SPECIFICATIONS AMENDMENT REQUEST

The proposed changes to Technical Specification sections 1.0, 2.0, 3/4.1.1, 3/4.1.2 and 3/4.10.1 reduce the boration requirements when shut down, by modification of the shutdown margin requirements as follows:

1. A new parameter, K_{N-1}, in introduced and defined as the K_{eff} calculated assuming the partially or fully inserted rod of highest inserted worth is fully withdrawn.

2. Limiting Conditions for Operation (LCOs) 3.1.1.1 and 3.1.1.2 require that for Modes 1-4, the shutdown margin be greater than or equal to 6% delta K/K, and for Mode 5, greater than or equal to 4% delta K/K. The proposed changes revise the shutdown margin requirements for Modes 1-5 according to full length Control Element Assembly (CEA) position. The revised Tech Spec 3.1.1.1 is applicable when all full length CEAs are fully inserted and requires that for Modes 3-5, the shutdown margin be greater than or equal to 1% delta K/K. The revised Tech Spec 3.1.1.2 is applicable when any full length CEA is withdrawn and requires that for Modes 1-5, the shutdown margin be greater than or equal to that given in a new Figure 3.1-1A. For reactor coolant cold leg temperature less than or equal to 500°F, K_{N-1} shall be less than 0.99. The LCO action statements are also revised to require boration when the above shutdown margin requirements are not met.

   Surveillance Requirements 4.1.1.1.1 and 4.1.1.2 require that the shutdown margin be verified at given time intervals to satisfy the LCO requirements. The proposed changes revise Tech Spec 4.1.1.1.1 and 4.1.1.2 to require the shutdown margin to be verified to the proposed new LCO requirements, as described above. In addition, the proposed change requires K_{N-1} to be determined to be less than 0.99 at least once every 24 hours.

   The associated Bases 3/4.1.1 and 3/4.1.2 are also revised to reflect the proposed changes.

3. The action statements for LCOs 3.1.2.2, 3.1.2.4, and 3.1.2.6, for Modes 1-4, require in part that when the requirements of the LCOs are not met, be in at least HOT STANDBY within the next 6 hours and borated to a shutdown margin equivalent to at least 6% delta K/K at 210 °F. The proposed changes delete the redundant reference to the shutdown margin requirement since these are covered in Tech Spec 3.1.1.1 and 3.1.1.2.

4. LCO 3.10.1 currently requires that a reactivity equivalent to at least the highest estimated CEA worth be available for trip insertion when the shutdown margin requirement of Tech Spec 3.1.1.1 is suspended for

measurement of CEA worth and shutdown margin during physics tests, and boration is required when that requirement is not met. The proposed change revises LCO 3.10.1 to state that the shutdown margin and K_N-1 requirements of Tech Spec 3.1.1.2 may be suspended for measurement of CEA worth and shutdown margin, provided reactivity equivalent to at least the highest estimated CEA worth is available for trip insertion from operable CEAs or the reactor is subcritical by at least the reactivity equivalent of the highest CEA worth.

5. The proposed change adds Special Test Exception 3.10.9 to allow the facility to suspend the requirements of Tech Spec 3.1.1.1 and 3.1.1.2 for the demonstration of the operability of the control element drive mechanism system during pre-startup tests. Basis 3/4.10.9 is also added to reflect this.

6. As a result of proposed changes 1-5 above, it is necessary to propose changes to revise setpoints to provide reactor trips to prevent the core from exceeding its safety limits in terms of departure from nucleate boiling ratio (DNBR) and local power density. These proposed changes consist of two parts:

   a. Item B.2 of Table 2.2-1 specifies a trip setpoint for "Excore Neutron Flux - Logarithmic Power Level - High" of less than or equal to 0.798% of rated thermal power, and an allowable value of less than or equal to 0.815% of rated thermal power. The proposed change revises the trip setpoint and allowable value to 0.010% and 0.011% of rated thermal power, respectively.

   b. Table notation (c) of Table 3.3-1 and Table notation (5) of Table 2.2-1 state that CPC trips may be manually bypassed below 1% of rated thermal power and the bypass shall automatically be removed when thermal power is greater than or equal to 1% of rated thermal power. The proposed changes revise the value at which the CPC trip may be manually bypassed and at which the manual bypass is automatically removed, from 1% of rated thermal power to 10^{-4}% of rated thermal power.

7. The proposed change renumbers Special Test Exception 3.10.9 "Natural Circulation Testing Program" (Unit 1 only) to 3.10.10. This is necessary so the Special Test Exception added in item 5 above will be the same section number in both Unit 1 and 2 Technical Specifications.

B. PURPOSE OF THE TECHNICAL SPECIFICATION

The purpose of the Technical Specifications affected by these proposed changes is to ensure that an adequate shutdown margin is maintained in the reactor at all times.

C. NEED FOR THE TECHNICAL SPECIFICATION AMENDMENT

This Technical Specification change will improve operational efficiency and provide ALARA benefits by reducing the amount of makeup water that must be processed during shutdown.

D. BASIS FOR PROPOSED NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

1. 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 the amendment request follows:

   Standard 1 -- Involve a significant increase in the probability or consequences of an accident previously evaluated.

   The proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated because the anticipated operational occurrences (AOOs) and accidents that have the potential for being impacted by the proposed changes have been reevaluated to determine the resulting consequences due to the proposed changes. These AOOs and accidents are steam line break, CEA withdrawal, CEA ejection, inadvertent deboration and startup of an inactive reactor coolant pump (RCP). The results of these evaluations show that the consequences are still within the appropriate acceptance criteria discussed below.

   The PVNGS safety analysis requires that steam line break events be evaluated considering potential for fuel damage. If the minimum DNBR during a steam line break event falls below specified limits based on acceptable correlations, fuel damage must be assumed. The results of the limiting steam line break analysis indicate that the minimum post-trip DNBR remains well above the specified safety limit.

   The PVNGS safety analysis requires that the consequences of an uncontrolled control element assembly (CEA) withdrawal from a subcritical or low power startup condition be evaluated on the basis that they are acceptable if the minimum DNBR remains above specified limits based on acceptable correlations. The reevaluation of the limiting CEA withdrawal analysis indicates that the minimum DNBR will remain above the plant specific safety limit of 1.231.

   The PVNGS safety analysis requires that for a startup of an inactive RCP, fuel clad integrity should be maintained by ensuring that specified acceptable fuel design limits are not exceeded. The results of a limiting startup of an inactive RCP indicate that the reactor remains subcritical and the specified acceptable fuel design limits are not exceeded, thus maintaining fuel clad integrity.

   The PVNGS safety analysis requires that for a CEA ejection, the reactivity excursion should not result in a radially averaged enthalpy greater than 280 cal/gm at any axial location in any fuel rod. Reevaluation of the limiting CEA ejection accident concurrent with the introduction of the K_{N-1} requirement ensures that the safety analysis acceptance criterion will be met.

The proposed change to lower the setpoint of the high logarithmic power trip will provide the trip function earlier than the previous setpoint. This trip provides protection in the event of an inadvertent control element assembly (CEA) bank withdrawal from MODES 2 and 3 initial conditions with four reactor coolant pumps (RCPs) operating. The proposed change to lower the value of power below which the CPC trip can be bypassed and above which the manual bypass is automatically removed also provides added protection for an inadvertent CEA bank withdrawal postulated to occur in MODES 3, 4, or 5 with less than four reactor coolant pumps operating. If the reactor coolant pressure or temperature is outside the CPC wide range trip limits, a continuous reactor trip signal will be generated by all four CPC channels and an immediate reactor trip will terminate an inadvertent CEA bank withdrawal event before significant power is generated. The MODE 2 and 3, with four RCPs operating, and the MODE 3, 4, and 5, with less than four RCPs operating events have determined to be less limiting than the CEA bank withdrawal event presented in the Final Safety Analysis Report. Also, the proposed changes do not alter how the CPCs respond to design basis events.

Therefore, operation of the facility in accordance with the proposed changes will not involve a significant increase in the probability or consequences of an accident previously evaluated.

Standard 2 -- Create the possibility of a new or different kind of accident from any accident previously evaluated.

The proposed changes will not create the possibility of a new or different kind of accident from any accident previously evaluated because there is no change in the plant hardware or analysis method as a result of the proposed changes. Although some of the proposed changes will result in modification to the operating procedures and plant operation in the shutdown modes, operation of the facility in accordance with the proposed changes will not create the possibility of a new or different kind of accident from any accident previously evaluated.

Standard 3 -- Involve a significant reduction in a margin of safety.

Operation of the facility in accordance with the proposed changes may reduce in some way a safety margin, but where the results of the change are clearly within the acceptance criteria.

The PVNGS safety analysis requires that for an inadvertent boron dilution, a minimum time interval of 15 minutes for MODES 1 through 5 and 30 minutes for MODE 6 be available from the time an alarm makes the operator aware of unplanned boron dilution before a loss of shutdown margin occurs. The time to a complete loss of shutdown margin for the limiting inadvertent boron dilution is now 50 minutes compared to 95 minutes for the previously analyzed incident. However, an alarm will alert the operator of an unplanned boron dilution at least 15 minutes prior to a complete loss of shutdown margin.

Although a numerically smaller value of SHUTDOWN MARGIN may appear to result in a reduction in the safety margin, operation of the facility in accordance with the proposed changes does not involve a significant reduction in a margin of safety as a alarm is available in time to satisfy the safety analysis criterion.

The lower logarithmic power level trip setpoint and automatic removal of the CPC manual bypass at a lower power level result in an earlier reactor protective system actuation for the postulated transients, which involves no significant reduction in the margin of safety.

2. The proposed changes in parts 1, 3, 4, 5, and 7 match the guidance concerning the application of the standards for determining whether or not a significant hazards consideration exists (51 FR 7751) by the example:

   (i) A purely administrative change to Technical Specifications : for example, a change to achieve consistency throughout the Technical Specifications, correction of an error, or a change in nomenclature.

The proposed changes in parts 2, 3, 4, and 5 match the guidance of 51 FR 7751 by the example:

(iv) A change which either may result in some increase to the probability or consequences of a previously analyzed accident or may reduce in some way a safety margin, but where the results of the change are clearly within all acceptable criteria with respect to the system or component specified in the Standard Review Plan; for example, a change resulting from the application of a small refinement of a previously used calculation model or design method.

The proposed changes in part 6 match the guidance of 51 FR 7751 by the example:

(ii) A change that constitutes an additional limitation, restriction or control not presently included in the technical specifications: for example, a more stringent surveillance requirement.

E. SAFETY EVALUATION FOR THE AMENDMENT REQUEST

The proposed Technical Specification amendment will not increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the FSAR. The anticipated operational occurrences (AOOs) and accidents that have potential for being impacted by the proposed changes are steam line break, CEA withdrawal, CEA ejection, inadvertent deboration and startup of an inactive reactor coolant pump (RCP). All the impacted AOOs and accidents have been reevaluated to determine the resulting consequences due to the proposed changes. The results of these evaluations show that the consequences are still within the appropriate acceptance criteria discussed below.

1. The PVNGS safety analysis requires that steam line break events be evaluated considering potential for fuel damage. If the minimum DNBR during a steam line break event falls below specified limits based on acceptable correlations, fuel damage must be assumed. The results of the limiting steam line break analysis indicate that the minimum post-trip DNBR remains well above the specified safety limit.

2. The PVNGS safety analysis requires that the consequences of an uncontrolled control element assembly (CEA) withdrawal from a subcritical or low power startup condition be evaluated on the basis that they are acceptable if the minimum DNBR remains above specified limits based on

acceptable correlations. The reevaluation of the limiting CEA withdrawal analysis indicates that the minimum DNBR will remain above the plant specific safety limit of 1.231.

3. The PVNGS safety analysis requires that for a startup or an inactive RCP, fuel clad integrity should be maintained by ensuring that specified acceptable fuel design limits are not exceeded. The result of a limiting startup of an inactive RCP indicate that the reactor remains subcritical and the specified acceptable fuel design limits are not exceeded, thus maintaining fuel clad integrity.

4. The PVNGS safety analysis requires that for a CEA ejection, the reactivity excursion should not result in a radially averaged enthalpy greater than 280 cal/gm at any axial location in any fuel rod. Reevaluation of limiting CEA ejection accident concurrent with the introduction of the K_{N-1} requirement ensures that the safety analysis acceptance criterion will be met.

The proposed change to lower the setpoint of the high logarithmic power trip will provide the trip function earlier than the previous setpoint. This trip provides protection in the event of an inadvertent control element assembly (CEA) bank withdrawal from MODES 2 and 3 initial conditions with four reactor coolant pumps (RCPs) operating. The proposed change to lower the value of power below which the CPC trip can be bypassed and above which the manual bypass is automatically removed also provides added protection for an inadvertent CEA bank withdrawal postulated to occur in MODES 3, 4, or 5 with less than four reactor coolant pumps operating. If the reactor coolant pressure or temperature is outside the CPC wide range trip limits, a continuous reactor trip signal will be generated by all four CPC channels and an immediate reactor trip will terminate an inadvertent CEA bank withdrawal event before significant power is generated. The MODE 2 and 3, with four RCPs operating, and the MODE 3, 4, and 5, with less than four RCPs operating events have been determined to be less limiting than the CEA bank withdrawal event presented in the Final Safety Analysis Report. Also, the proposed changes do not alter how the CPCs respond to design basis events.

Therefore, operation of the facility in accordance with the proposed changes will not involve a significant increase in the probability or consequences of an accident previously evaluated.

The proposed Technical Specification amendment will not create the possibility for an accident or malfunction of a different type than any previously evaluated in the FSAR. There is no change in the plant hardware or analysis method as a result of the proposed changes. Although some of the proposed changes will result in modification to the operating procedures and plant operation in the shutdown modes, operation of the facility in accordance with the proposed changes will not create the possibility of a new or different kind of accident from any accident previously evaluated.

Operation of the facility in accordance with the proposed changes may reduce in some way a safety margin, but where the results of the change are clearly within the acceptance criteria.

The PVNGS safety analysis requires that for an inadvertent boron dilution, a minimum time interval of 15 minutes for MODES 1 through 5 and 30 minutes for MODE 6 be available from the time an alarm makes the operator aware of unplanned boron dilution before a loss of shutdown margin occurs. The time to complete loss of shutdown margin for the limiting inadvertent boron dilution is now 50 minutes compared to 95 minutes for the previously analyzed incident. However, an alarm will alert the operator of an unplanned boron dilution at least 15 minutes prior to a complete loss of shutdown margin.

Although a numerically smaller value of SHUTDOWN MARGIN may appear to result in a reduction in the safety margin, operation of the facility in accordance with the proposed changes does not involve a significant reduction in a margin of safety as an alarm is available in time to satisfy the safety analysis criterion.

The lower logarithmic power level trip setpoint and automatic removal of the CPC manual bypass at a lower power level result in an earlier reactor protective system actuation for the postulated transients, which involves no significant reduction in the margin of safety.

F. ENVIRONMENTAL IMPACT CONSIDERATION DETERMINATION

The proposed change request does not involve an unreviewed environmental question because operation of PVNGS Units 1 and 2, in accordance with this change, would not:

1. Result in a significant increase in any adverse environmental impact previously evaluated in the Final environmental statement (FES) as modified by the staff's testimony to the Atomic Safety and Licensing Board, Supplements to the FES, Environmental Impact appraisals, or in any decisions of the Atomic Safety and Licensing Board; or

2. Result in a significant change in effluents or power levels; or

3. Result in matters not previously reviewed in the licensing basis for PVNGS which may have a significant environmental impact.

G. MARKED-UP TECHNICAL SPECIFICATION CHANGE PAGES:

Index: I, II, IV, IX, X, XIV

Definitions:

1-4, 1-5, 1-6, 1-7

Safety Limits and Safety System Settings:

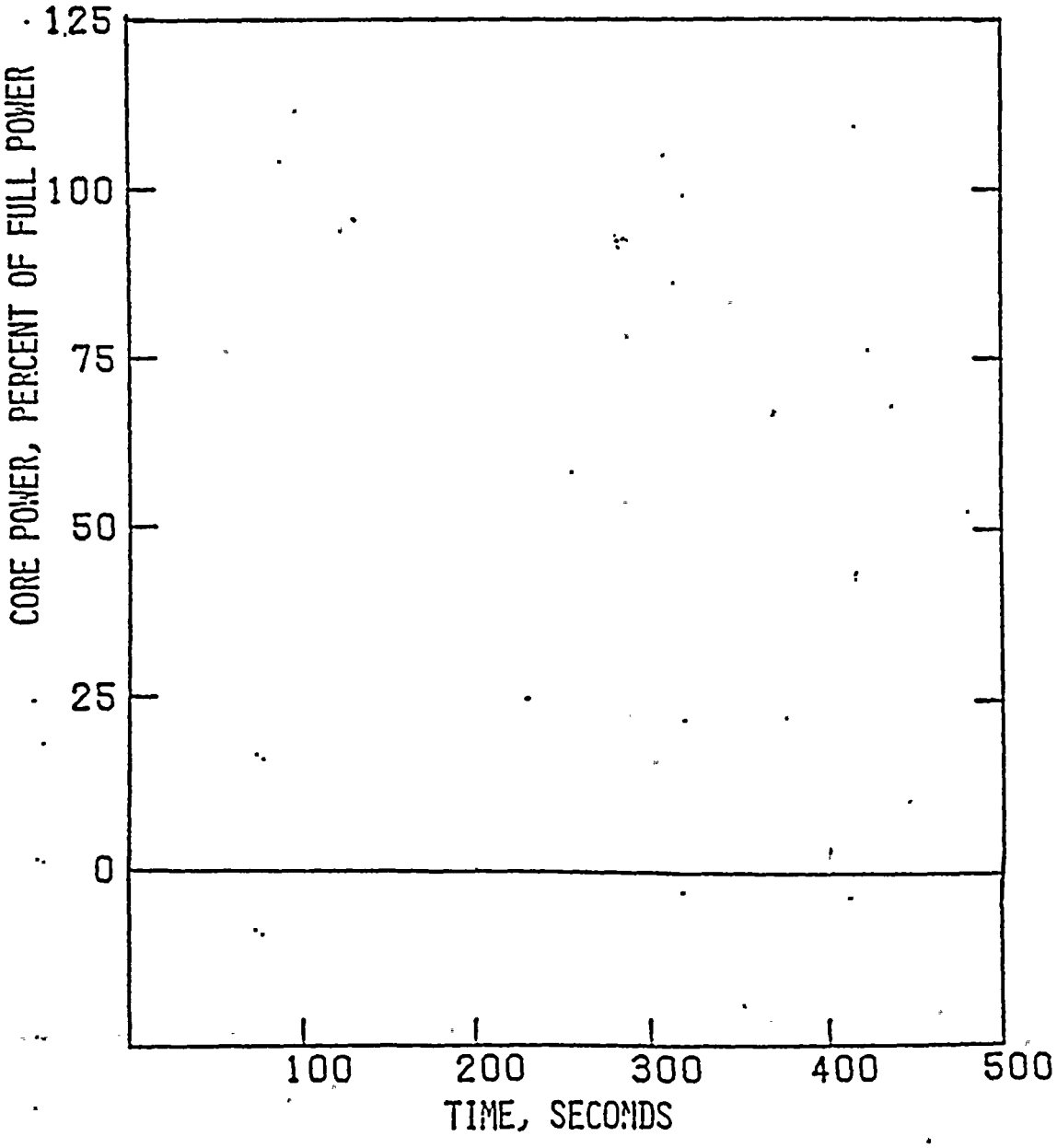
2-4, 2-5

Limiting Conditions for Operation and Surveillance Requirements:

3/4 1-1
1-2
1-2a
1-3
1-8
1-10
1-13
3-5
10-1
10-9
10-10 (Unit 1 only)

Bases for Limiting Conditions for Operation and Surveillance Requirements

B 3/4 1-1
1-1a
1-2
10-2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75
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25
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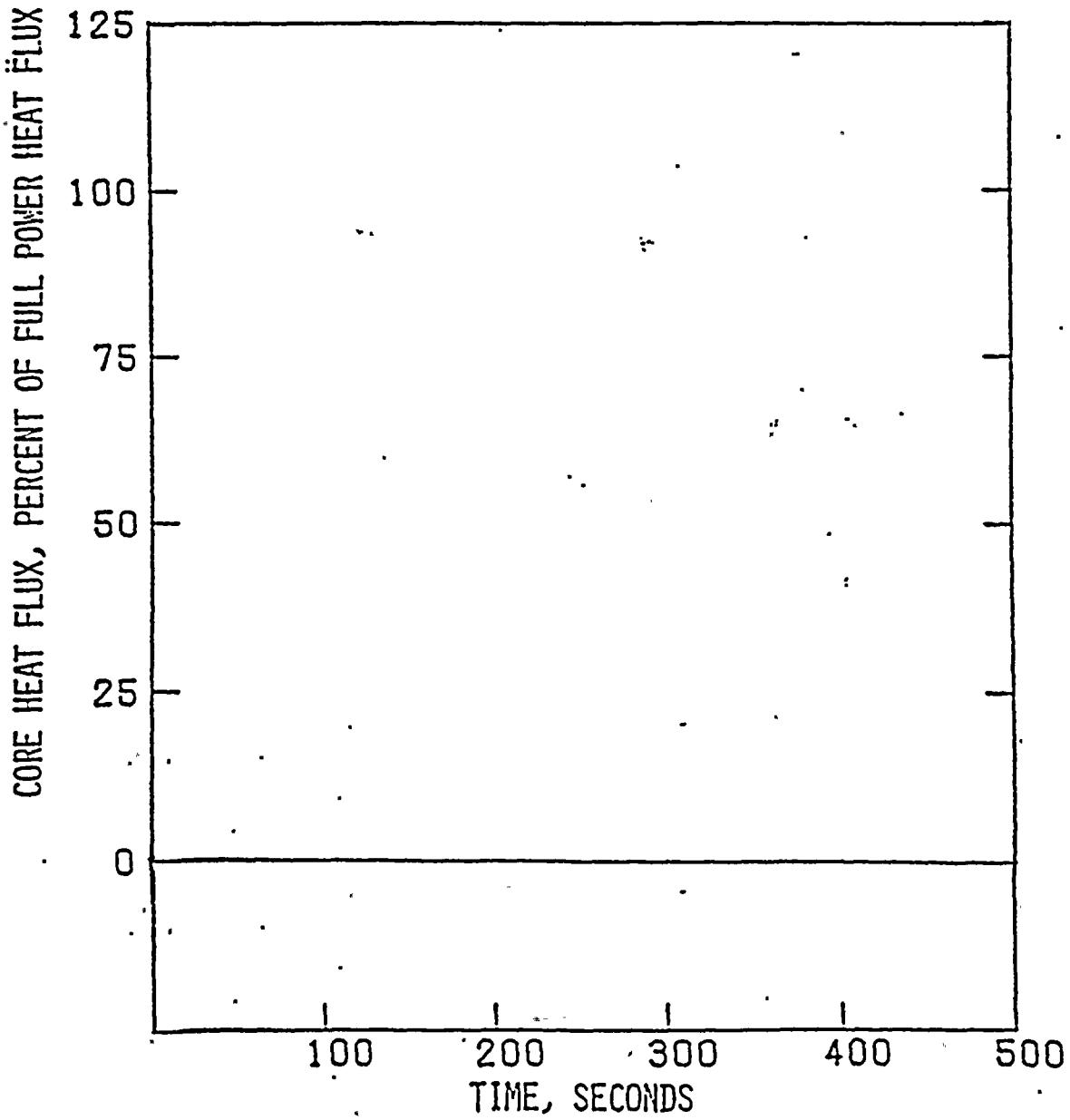
CORE POWER, PERCENT OF FULL POWER
100
200
300
400
500
TIME, SECONDS

Figure 1-1


CORE HEAT FLUX, PERCENT OF FULL POWER HEAT FLUX
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TIME, SECONDS

Figure 1-2

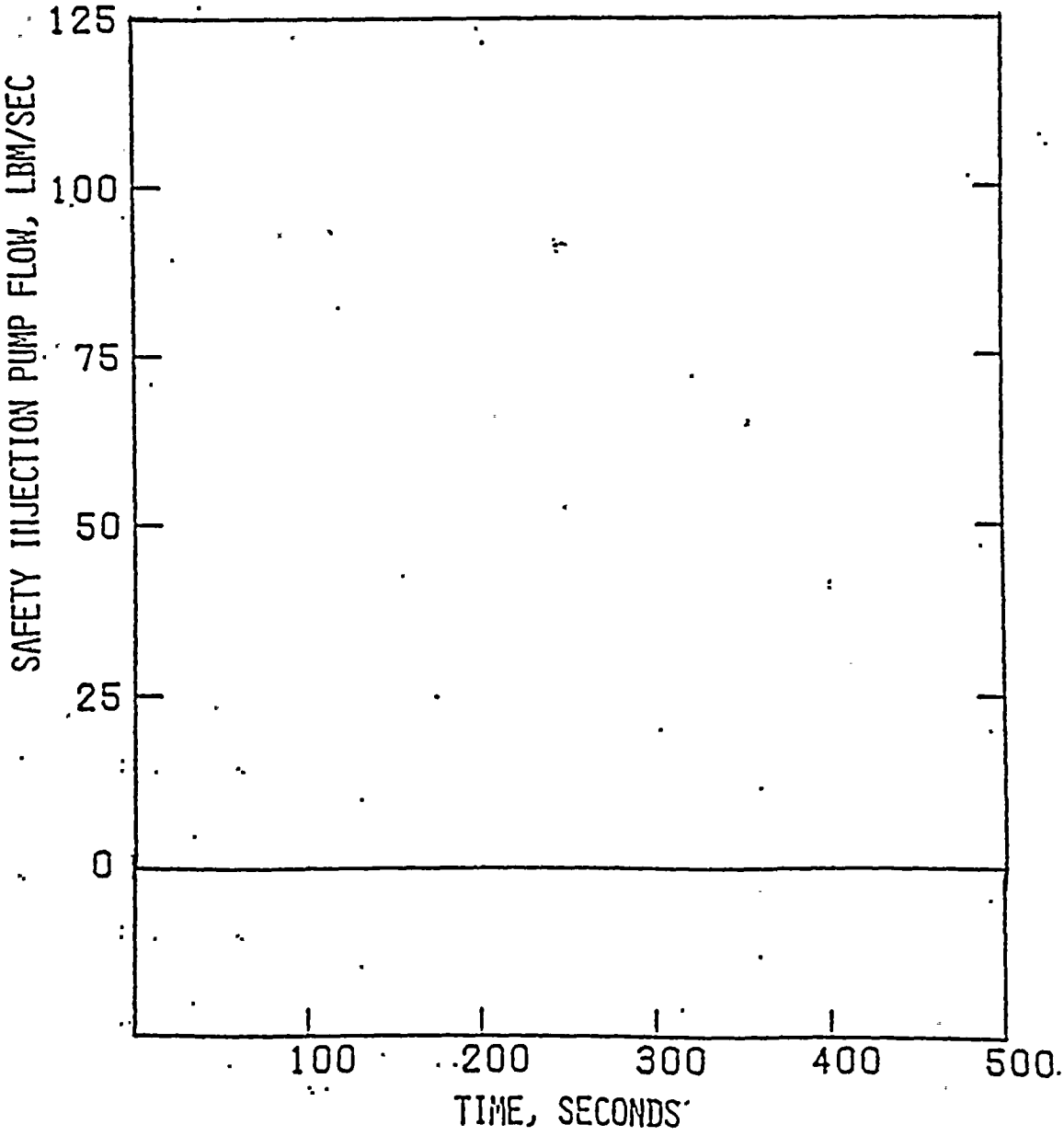
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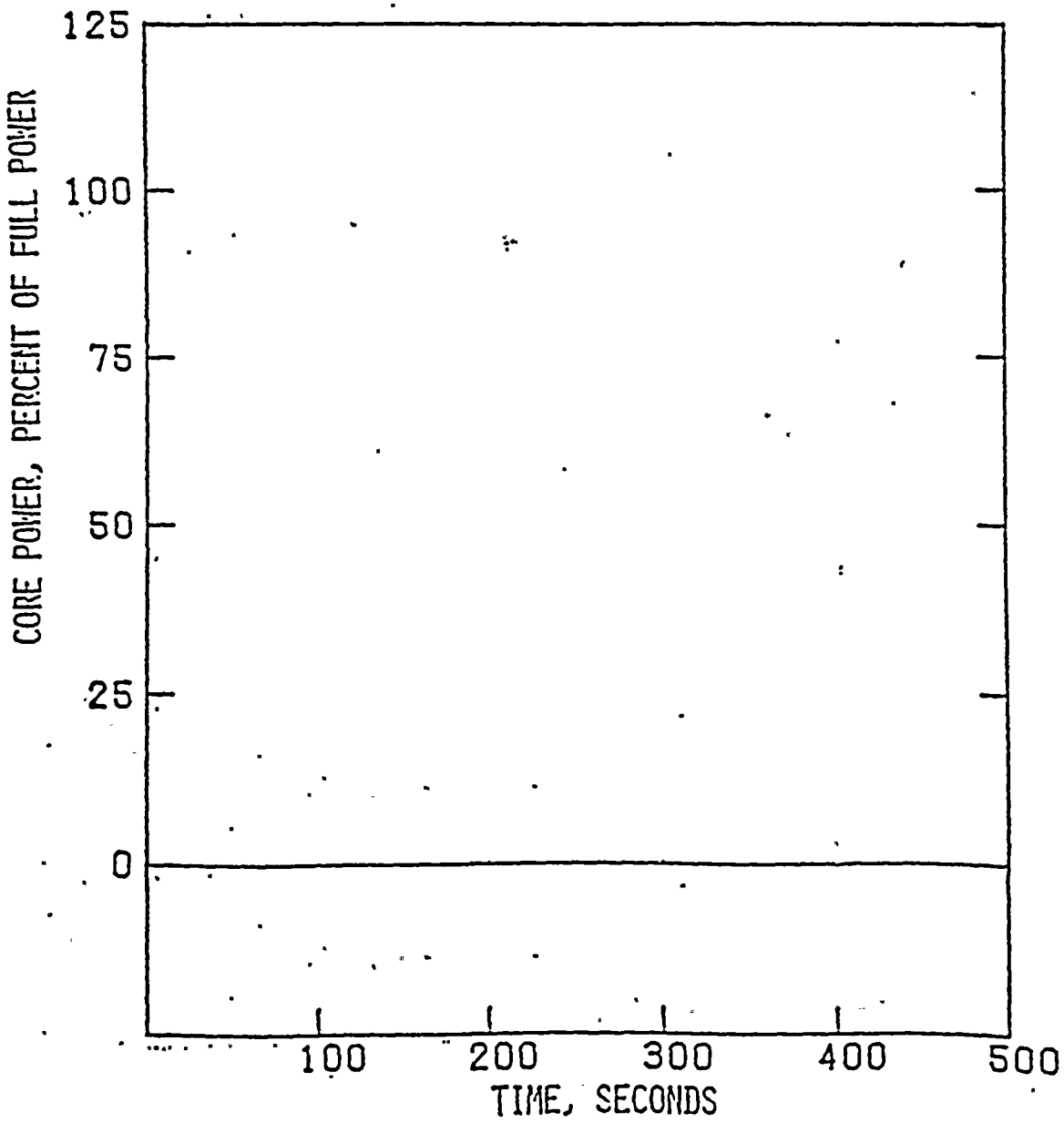
Figure 1-15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CORE POWER, PERCENT OF FULL POWER
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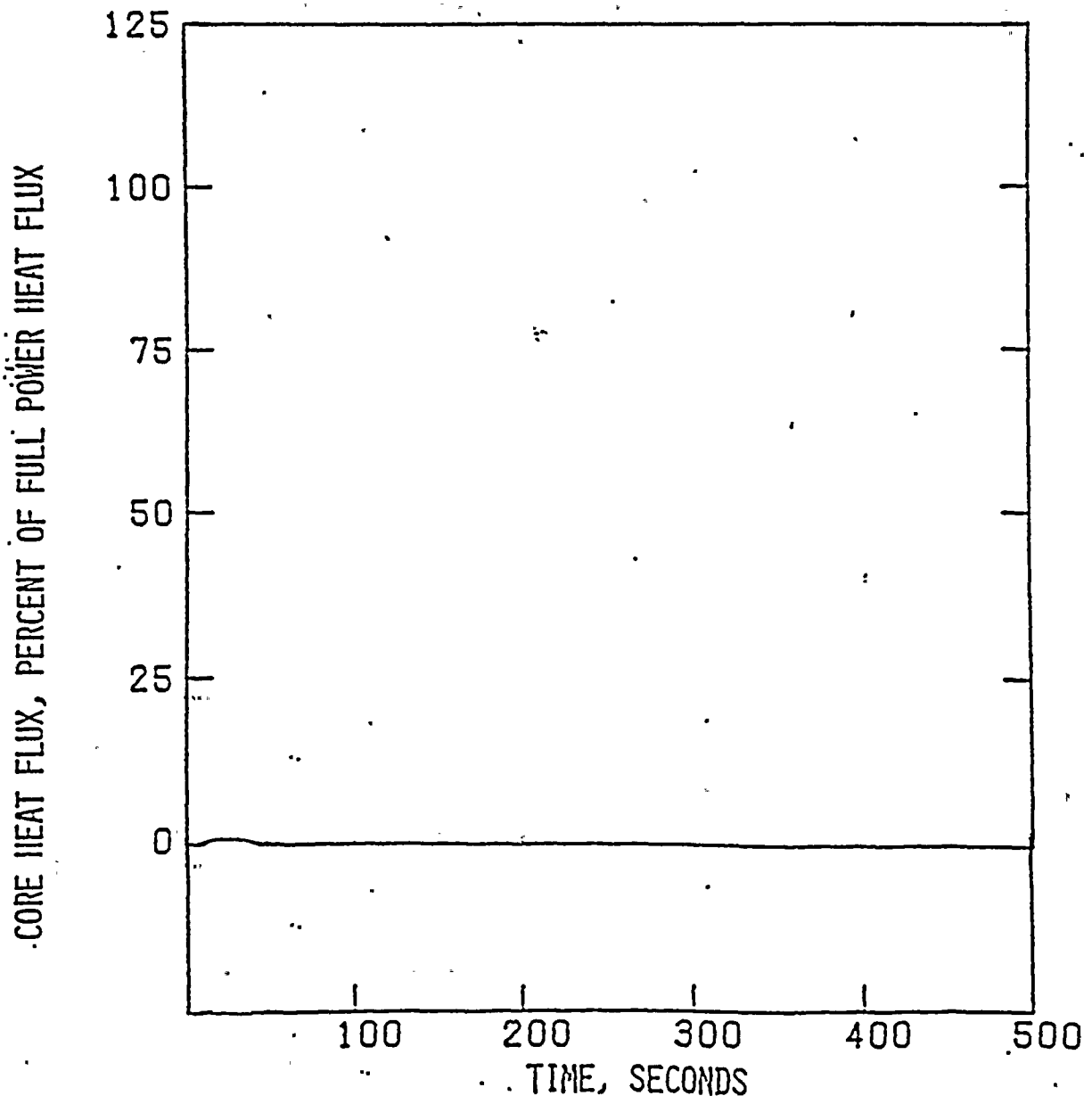
TIME, SECONDS

Figure 1-17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CORE HEAT FLUX, PERCENT OF FULL POWER HEAT FLUX
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300
400
500
TIME, SECONDS

Figure 1-18

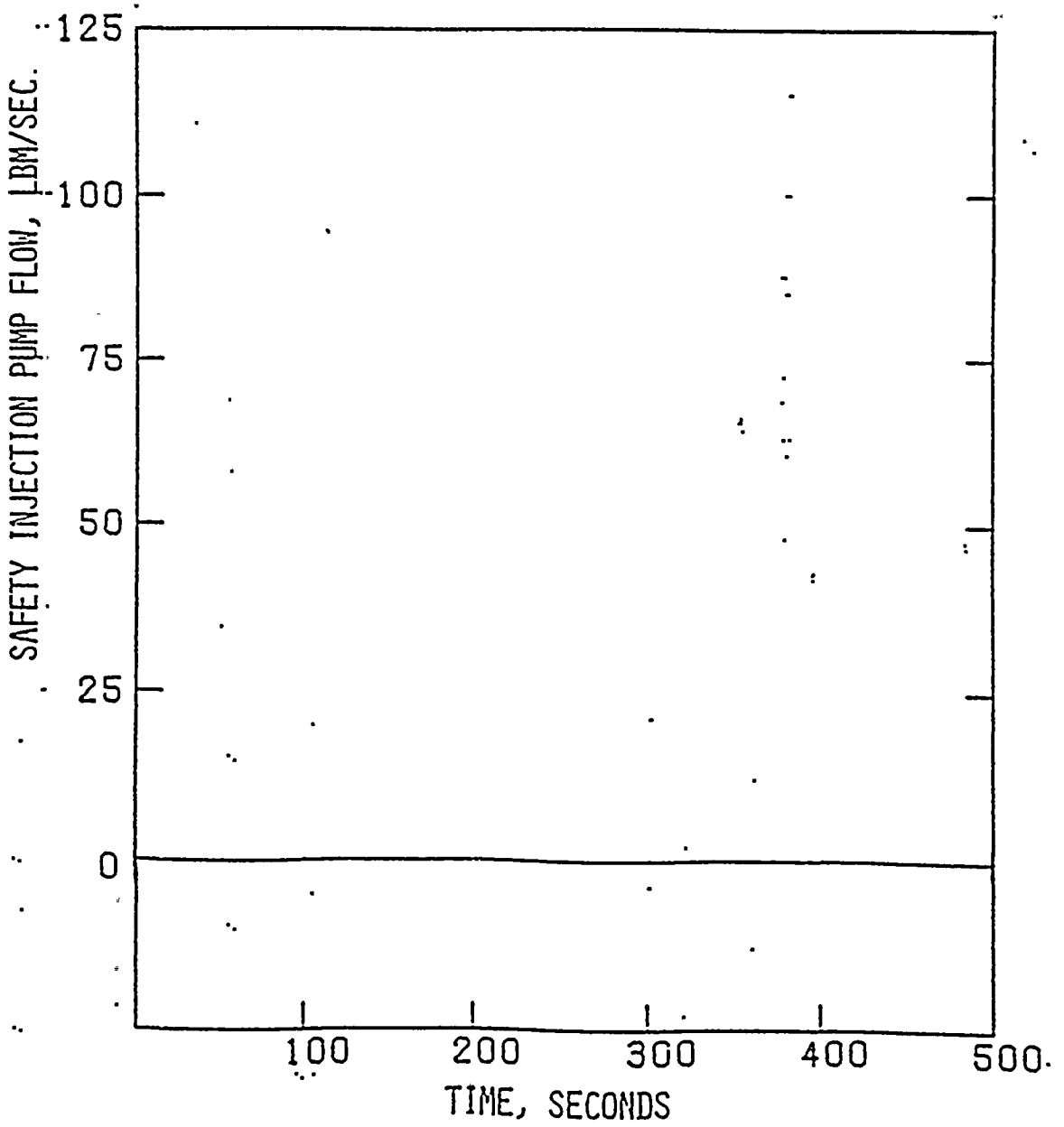
125
100
75
50
25
0
SAFETY INJECTION PUMP FLOW, LBM/SEC.
100
200
300
400
500.
TIME, SECONDS

Figure 1-31