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Project Number 691

BWROG-04021
August 5, 2004

US Nuclear Regulatory Commission
Document Control Desk
Washington, D.C. 20852-2738

SUBJECT: **NRC ACCEPTED VERSION OF IMPROVED BPWS CONTROL ROD INSERTION PROCESS LTR, NEDO-33091-A**

Reference: 1. USNRC, "Safety Evaluation For Licensing Topical Report (LTR) NEDO-33091, "Improved BPWS Control Rod Insertion Process" (TAC No. MB9642)

Attachment: BWR Owners' Group Licensing Topical Report, *Improved BPWS Control Rod Insertion Process*, NEDO-33091-A (non-proprietary), Revision 2, July 2004.

In accordance with the Reference 1, this letter transmits (attached) the NRC accepted version of the subject LTR. Per Reference 1, the attached LTR version incorporates the NRC cover letter and the Safety Evaluation between the title page and the abstract. The Table of Contents has been updated such that new information can be readily located. This version has added two appendices of historical review information (the associated NRC Requests for Additional Information and their accepted responses, and the BWROG comments on the draft NRC SE). All of the original report pages have been replaced with pages that include an "-A" in the report identification number.

Very truly yours,

Kenneth S Putnam

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D044

GE Nuclear Energy

NEDO-33091-A
Revision 2
Class I
DRF 0000-0008-3745
July 2004

NRC Accepted

BWR Owners' Group Licensing Topical Report

Improved BPWS
Control Rod Insertion Process

GE Nuclear Energy

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NEDO-33091-A
Revision 2
Class I
DRF 0000-0008-3745
July 2004

NRC Accepted

BWR Owners' Group
Licensing Topical Report

Improved BPWS Control Rod Insertion Process

Approved by: J. Tuttle, Project Manager

IMPORTANT NOTICE REGARDING THE CONTENTS OF THIS REPORT

Please Read Carefully

DISCLAIMER

CHANGES FROM PREVIOUS VERSION

In accordance with the NRC cover letter to the SAFETY EVALUATION FOR LICENSING TOPICAL REPORT (LTR) NEDO-33091, "IMPROVED BPWS CONTROL ROD INSERTION PROCESS," (TAC No. MB9642), this report is the (NRC) accepted version of the LTR. This version incorporates the NRC cover letter and the Safety Evaluation between the title page and the abstract. The Table of Contents has been updated such that new information can be readily located. This version has added two appendices of historical review information (the associated NRC Requests for Additional Information and their accepted responses, and the BWROG comments on the draft NRC SE). All of the original report pages have been replaced with pages that include a "-A" in the report identification number. The technical content in the body of this LTR has not changed, and thus, the revision number has not changed.

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

June 16, 2004

Mr. Kenneth Putnam, Chairman
BWR Owners Group
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Duane Arnold Energy Center
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SUBJECT: SAFETY EVALUATION FOR LICENSING TOPICAL REPORT (LTR) NEDO-33091, "IMPROVED BPWS CONTROL ROD INSERTION PROCESS" (TAC NO. MB9642)

Dear Mr. Putnam:

On June 6, 2003, and supplemented on April 21, 2004, the Boiling Water Reactors Owners Group (BWROG) submitted LTR NEDO-33091, "Improved BPWS Control Rod Insertion Process," to the staff for review and approval. On May 10, 2004, an NRC draft safety evaluation (SE) regarding our approval of NEDO-33091 was provided for your review and comments. By telecon on June 3, 2004, the BWROG provided minor editorial comments. The staff has incorporated the BWROG's comments into the final SE enclosed with this letter.

The staff has found LTR NEDO-33091 acceptable for referencing in licensing applications for boiling water reactors to the extent specified and under the limitations delineated in the LTR and in the enclosed SE. The SE defines the basis for acceptance of the LTR.

Our acceptance applies only to matters approved in the subject LTR. We do not intend to repeat our review of the acceptable matters described in the LTR. When the LTR appears as a reference in license applications, our review will ensure that the material presented applies to the specific plant involved. License amendment requests that deviate from this LTR will be subject to a plant-specific review in accordance with applicable review standards.

In accordance with the guidance provided on the NRC web site, we request that the BWROG publish an accepted version of this LTR within three months of receipt of this letter. The accepted version shall incorporate this letter and the enclosed SE between the title page and the abstract. It must be well indexed such that information is readily located. Also, it must contain in appendices historical review information, such as questions and accepted responses, draft SE comments, and original report pages that were replaced. The accepted version shall include a "-A" (designated accepted) following the report identification symbol.

If the NRC's criteria or regulations change so that its conclusion in this letter, that the LTR is acceptable, are invalidated, the BWROG and/or the applicant referencing the LTR will be expected to revise and resubmit its respective documentation, or submit justification for the continued applicability of the LTR without revision of the respective documentation.

Sincerely,

Herbert N. Berkow, Director
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Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Project No. 691

Enclosure: Safety Evaluation

cc w/encl: See next page

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

LICENSING TOPICAL REPORT NEDO-33091, "IMPROVED BPWS CONTROL ROD INSERTION PROCESS"

BOILING WATER REACTOR OWNERS GROUP (BWROG)

PROJECT NO. 691

1.0 INTRODUCTION

By letter dated June 6, 2003, as supplemented by letter dated April 21, 2004, the BWROG requested the NRC to review its licensing topical report (TR) NEDO-33091, "Improved BPWS [Banked Position Withdraw Sequence] Control Rod Insertion Process." Both the original BPWS process previously approved by the staff and the proposed improved process, are designed to minimize reactivity insertion during a postulated design basis control rod drop accident (CRDA).

Throughout its operating cycle, a boiling water reactor (BWR) experiences various startup, normal, and shutdown operations. Control rods are also moved due to fuel burn-up, power maneuvers, and normal operational occurrences. This rod movement could potentially result in a decoupled control rod that's stuck in the core, followed by a subsequent control rod drop, which would lead to a high reactivity insertion in a small region of the core. For large loosely coupled cores, a significant shift in the spatial power generation could occur during the course of this excursion. Utilizing rod pattern control systems, i.e., rod worth minimizer, rod sequence control system or rod pattern controller, the BPWS was developed to reduce the maximum control rod worth during the startup and shutdown processes. The original/standard BPWS process currently requires control rods to be moved in banked positions, even during the shutdown process after the low power set point (LPSP) is reached. This requirement results in the control of rod movement through many steps, when there is an extremely low possibility for the control rod to drop out of the core. Therefore, the improved BPWS proposes the one-step full insertion of control rods without banking after the reactor power is below LPSP.

2.0 REGULATORY BASIS

CRDA is the design basis accident for the subject LTR. In order to minimize the impact of a CRDA, the BPWS process was developed to minimize control rod reactivity worth for BWR2-6. The proposed improved BPWS further simplifies the control rod insertion process, and in order to evaluate it, the staff followed the guidelines of Standard Review Plan Section 15.4.9, and referred to General Design Criterion 28 of Appendix A to 10 CFR Part 50 as its regulatory requirement.

3.0 TECHNICAL EVALUATION

The original/standard BPWS was developed to minimize the control rod worth and mitigate the consequences of a CRDA from occurring during startup. This procedure also directly applies to the control rod insertion sequence during the shutdown routine, after power is lower than the LPSP. The BWROG and GE Nuclear Energy (GENE) found that this approach, while conservative, requires unnecessary control rod movements during the shutdown process. The procedural requirements on the operator also increases the risk of incorrect control rod movement, and causes additional wear on the rod and rod drive hardware systems. Since the possibility of having a decoupled control rod is extremely low during the shutdown process, GENE is proposing the improved BPWS, which allows control rods to be fully inserted in a single step during the shutdown process.

The improved BPWS proposes the following changes to the operational procedures:

1. Before reducing power to the LPSP, operators shall confirm control rod coupling integrity for all rods that are fully withdrawn. Control rods that have not been confirmed coupled and are in intermediate positions must be fully inserted prior to power reduction to the LPSP. No action is required for fully-inserted control rods.

   If a shutdown is required and all rods, which are not confirmed coupled, cannot be fully inserted prior to the power dropping below the LPSP, then the original/standard BPWS must be adhered to.

2. After reactor power drops below the LPSP, rods may be inserted from notch position 48 to notch position 00 without stopping at intermediate positions. However, GENE recommends that operators should insert rods in the same order as specified for the original/standard BPWS as much as reasonably possible. If a plant is in the process of shutting down following improved BPWS with the power below the LPSP, no control rod shall be withdrawn unless the control rod pattern is in compliance with standard BPWS requirements.

All other control rod operational requirements are unchanged and continue to apply. The proposed changes may alter the technical specifications of certain plants; GENE has identified the potentially affected areas in the standard technical specifications. The specific changes for each plant implementing the improved BPWS will be determined on a case-by-case basis.

The basis of the improved BPWS is the assumption that a CRDA can only be caused by a stuck rod which is decoupled from the control rod drive (CRD). No single failure of a BWR CRD mechanical or hydraulic system can cause a control rod to drop completely out of the reactor core during the reactor shut-down process. In its April 21, 2004, response to the staff's request for additional information (RAI), the BWROG/GENE referred the staff to Final Safety Analysis Report (FSAR) sections, isometric drawings, and hydraulic schematics describing the CRD hydraulic unit design, control rod assembly configuration, and postulated CRD failure modes and effects scenarios from the FSARs for Oyster Creek (BWR/2), Monticello (BWR/3), Limerick (BWR/4), LaSalle (BWR/5), and Perry (BWR/6). The staff's review considered CRD hydraulic systems from plants of various BWR designs, and found that the CRD systems of BWR/2 through BWR/6 designs are very similar with respect to the mechanisms for rod insertion, withdrawal, and locking. The staff found that during a reactor shutdown process for all operating BWRs when each control rod is given an insert signal, there exists no single failure of the CRD hydraulic or mechanical system that could result in a control rod withdrawal out of the core of more than six inches (equivalent to one CRD index tube drive notch length). Therefore, the staff agrees with the BWROG/GENE's assessment regarding the possible cause of a CRDA during the shutdown process after reactor power is below the LPSP since the technical basis, as cited above, is sound and acceptable.

Implementation of the improved BPWS requires two major operating procedure changes. The requirement for operators to confirm control rod coupling integrity for all rods fully withdrawn will ensure proper coupling during the control rod insertion process and any possible rod withdrawal after reactor power drops below LPSP. The proposed procedure for the full insertion of all unconfirmed control rods prior to LPSP will prevent the possibility of a decoupled control rod dropping out during the control rod maneuvers. If all unconfirmed control rods cannot be fully inserted prior to the LPSP, the use of the standard BPWS will become the conservative fall back position, since the risk of a CRDA occurring using the improved BPWS will be no different than the original/standard BPWS using this procedure.

After reactor power drops below the LPSP, the improved BPWS allows the full insertion of each control rod without banking. This simplification of the control rod insertion process helps to reduce the number of control rod insertion steps. Since all unconfirmed control rods have been inserted, it is highly unlikely for a CRDA to occur while confirmed rods are being inserted without banking. Therefore, the improved BPWS will have the same level of safety assurance as the previously approved standard BPWS process. Should the operator decide to reverse the shutdown process, the improved BPWS does not allow for the withdrawal of any control rods, unless the control rod pattern meets the standard BPWS requirements. This ensures that all control rods are always banked for withdrawal.

The improved BPWS's single step full insertion also reduces the insertion time of each rod, which may induce a necessary increase in other procedures or processes to accommodate this rapid change. During telephone conferences, the staff requested additional information from the BWROG/GENE regarding the impact of the accelerated shut-down process on other procedures. The BWROG/GENE examined its process and requirements, and concluded in its RAI response on April 21, 2004, that the improved BPWS process does not adversely affect the normal shutdown processes, since the operating procedures will remain to be bounded by the most limiting (fastest negative reactivity) control rod insertion scenario (RAI #3). In addition, pressure-temperature effects, as in the cooldown process for example, are accounted for and controlled by controlling reactor dome pressure, coolant flow and coolant temperature.

4.0 CONCLUSIONS

The BWROG/GENE has proposed an improved BPWS process which allows for the single step full insertion of control rods during shutdown, when the reactor power is lower than the LPSP. The staff has completed its review of the subject LTR, and concluded that the proposed change is acceptable and applicable to BWR/2-6 with original/standard BPWS already implemented.

Plants electing to implement the improved BPWS must reflect the changes in their operating procedure. If the technical specification of a plant is impacted or needs to be updated, an amendment submittal to the NRC will be required.

Principal Contributor: Shanlai Lu

Date: June 16, 2004

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ACRONYMS AND ABBREVIATIONS

APRM	Average Power Range Monitor
BPWS	Banked Position Withdrawal Sequence
BWR	Boiling Water Reactor
BWROG	Boiling Water Reactor Owners Group
CRD	Control Rod Drive
CRDA	Control Rod Drop Accident
IRM	Intermediate Range Monitor
LCO	Limiting Condition of Operation
LPSP	Low Power Setpoint
LTR	Licensing Topical Report
NRC	Nuclear Regulatory Commission
OLTP	Original Licensed Thermal Power
RAI	Request for Additional Information
RPC	Rod Pattern Controller
RPCS	Rod Pattern Control System
RSCS	Rod Sequence Control System
RTP	Rated Thermal Power
RWM	Rod Worth Minimizer
SDM	Shutdown margin
SOE	Single Operator Error
STS	Standard Technical Specifications
TS	Technical Specifications
TSTF	Technical Specification Task Force

EXECUTIVE SUMMARY

This generic Licensing Topical Report (LTR) presents an improved Banked Position Withdrawal Sequence (BPWS) for performing reactor shutdowns. This report justifies modifying the requirements of the Standard Technical Specifications (STS) relative to the applicability of the systems used to adhere to the BPWS during the reactor shutdown process (i.e., control rods are specifically being inserted to achieve shutdown at a power level less than the low power setpoint (LPSP)). The proposed improvement to the reactor shutdown process allows each control rod to be fully inserted to position 00 in one step instead of banking (e.g., 48-12-8-4-00) below the LPSP. To utilize this version of the BPWS process, it is required that control rods that have not been confirmed to be coupled, are fully inserted prior to reducing power below the LPSP. The BPWS control rod groups are unchanged.

The BPWS, as currently implemented, limits the potential reactivity increase from a postulated Control Rod Drop Accident (CRDA) during reactor startups and shutdowns below the LPSP (generically based on 10% of original licensed thermal power). During the reactor shutdown process, confirming that control rods are coupled prior to decreasing power below the LPSP eliminates the postulated scenario for a CRDA, and thus, the CRDA would no longer be a credible event.

Modifying plant Technical Specifications (TS) and/or their Bases to reflect the use of the improved BPWS process would allow control rods to be fully inserted in a single step during the reactor shutdown process below the LPSP. This provides the following benefits:

- Allows the plant to reach the all-rods-in condition prior to significant reactor cool down, which reduces the potential for a re-criticality as the reactor cools down;
- Reduces the potential for an operator reactivity control error by reducing the total number of control rod manipulations;
- Minimizes the need for manual scrams during plant shutdowns, resulting in less wear on Control Rod Drive (CRD) system components and CRD mechanisms; and
- Eliminates unnecessary control rod manipulations at low power, resulting in less wear on Reactor Manual Control and CRD system components.

ACKNOWLEDGMENTS

The following individuals contributed significantly to the development, verification, and review of this report:

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1. BACKGROUND

The design basis reactivity insertion event for the BWR is the Control Rod Drop Accident (CRDA). From Section S.2.2.3.1 of Reference 1, the CRDA scenario postulates the following:

(a) Reactor is at a control rod pattern corresponding to maximum incremental rod worth.

(b) Rod pattern control systems (Rod Worth Minimizer, Rod Sequence Control System or Rod Pattern Controller) or operators are functioning within the constraints of the Banked Position Withdrawal Sequence (BPWS). The control rod that results in the maximum incremental reactivity worth addition at any time in core life under any operating condition while employing the BPWS becomes decoupled from the control rod drive.

(c) Operator selects and withdraws the drive of the decoupled rod along with the other required control rods assigned to the Banked–position group such that the proper core geometry for the maximum incremental rod worth exists.

(d) Decoupled control rod sticks in the fully inserted position.

(e) Control rod becomes unstuck and drops at the maximum velocity determined from experimental data (3.11 feet per second).

(f) Reactor goes on a positive period and initial power burst is terminated by the Doppler reactivity feedback.

(g) APRM 120% power signal scrams reactor (conservative; in startup mode APRM scram would be operative + IRM).

(h) Scram terminates accident.

The use of the BPWS ensures that no CRDA could exceed the applicable event limits, by reducing the incremental control rod reactivity worth to acceptable values.

The BPWS is described in detail in the Reference 2 LTR. The control rods are divided into 10 groups, with the first four groups representing approximately 50% of the control rods in a checkerboard pattern. During the reactor startup process, starting from the all-rods-in condition, the BPWS allows each control rod in the first 25% of the control rods (i.e., first two control rod groups) to be fully withdrawn (in a predetermined group sequence) from notch 00 to notch 48. The second 25% of the controls rods (i.e., second two control rod groups) to be withdrawn are then banked to notch positions 00→N_1→N_2→N_3→N_4→48, where all control rods within a group must be withdrawn to each designated bank position before proceeding to the next bank position (where N_i represents an intermediate notch position, e.g., 04, 08 or 12.). After 50% of the control rods are completely withdrawn, the remaining control rod groups are withdrawn in a similar manner until the reactor exceeds the LPSP (generically based on 10% of original licensed thermal power (OLTP)).

The CRDA is primarily of concern during reactor startups, because the act of withdrawing a control rod can cause the rod to become decoupled from its drive assembly. It is impossible to drop a coupled control rod or a coupled control rod that is in the process of being inserted.

During normal operations, routine control rod coupling checks are performed, and these ensure that the fully withdrawn control rods are coupled. During the shutdown process, for the withdrawn control rods that are confirmed to be coupled, the possibility of a CRDA is eliminated, and thus, banking withdrawn control rods in to the BPWS intermediate positions is not needed.

Predetermined control rod withdrawal sequences control the power distribution in the core, and minimize control rod worth. From the all-rods-in condition to the LPSP, either the Rod Pattern Controller (RPC), Rod Sequence Control System (RSCS), or the Rod Worth Minimizer (RWM) (depending on the plant design) enforces the BPWS constraints on control rod movements. Above the LPSP, inherent feedback mechanisms, primarily in the form of steam voids, limit the control rod worth such that a CRDA does not exceed the applicable event limits.

The BPWS is required by the STS to be applied to both reactor startup and shutdown processes. Because of the delay caused by the use of the Reference 2 version of the BPWS in achieving shutdown, some plants perform manual scrams instead of going through the multiple-step BPWS shutdown process (approximately 400 steps for a medium sized reactor).

2. INTRODUCTION

Reference 2 conservatively applies the BPWS intermediate steps to both startups and shutdowns, without regard to the fact that compensatory operator actions could eliminate the possibility of a CRDA during the reactor shutdown process. The improved BPWS control rod insertion process, described herein, provides the compensatory operator actions that allow control rods to be fully inserted in a single step.

This report addresses changes to the shutdown process which currently constrains the control rod insertion sequence. The proposed changes:

1. Require control rod coupling confirmations, which eliminate any Single Operator Error (SOE) with respect to assuring if the withdrawn control rods are coupled.
2. Require each control rod that has not been confirmed coupled (since its last withdrawal) to be fully inserted prior to reducing power below the LPSP. (These rods are usually partially inserted rods at high power.)
3. Allow each remaining (i.e., coupled) control rod to be fully inserted in a single step below the LPSP, instead of requiring each control rod to be banked at intermediate positions. (For some plants, this requires the Rod Worth Minimizer (RWM) or Rod Pattern Controller (RPC) to be bypassed.)

All other control rod operability requirements are unchanged and continue to apply. Allowing each control rod to be fully inserted in a single step reduces the total rod manipulation steps to shutdown a reactor from ~400 for a medium sized reactor to ~150 steps. This reduction would result in:

- Less chance of a re-criticality as the reactor cools down,
- Reducing the potential for operator errors,
- Fewer manual scrams and less wear on control and CRD system components, and
- Eliminates unnecessary control rod manipulations.

In this report, Section 3 addresses the current BPWS, RWM and RPC requirements in the Standard Technical Specifications (STS), Section 4 provides the technical justification for the elimination of the intermediate (banked) steps of the BPWS during the reactor shutdown process, Section 5 provides guidance for plant procedural checks, Section 6 provides proposed STS changes, and Section 7 discusses the effects on plant equipment and benefits.

3. STANDARD TECHNICAL SPECIFICATIONS ADDRESSING BPWS AND RWM/RPC

This section summarizes the current Standard Technical Specifications (STS) with respect to the use of the BPWS, and RWM (or RPC for a BWR/6). Generic examples of the requirements for applicability of the BPWS and RWM/RPC are contained in the BWR/4 STS (NUREG 1433 - Reference 3) and BWR/6 STS (NUREG 1434 - Reference 4). The potentially affected STS locations are listed below.

BWR/4 NUREG 1433 Locations	**BWR/6 NUREG 1434 Locations**
STS 3.1.3; CONDITION D and REQUIRED ACTION D.1	STS 3.1.3; CONDITION D and REQUIRED ACTION D.1
STS 3.1.6; LCO 3.1.6, CONDITIONS A and B, REQUIRED ACTIONS A.1 and B.1, and SR 3.1.6.1	STS 3.1.6; LCO 3.1.6, CONDITIONS A and B, and SR 3.1.6.1
STS 3.3.2.1; CONDITION C, REQUIRED ACTIONS C.2.2 and D.1, and SR 3.3.2.1.8	STS Table 3.3.2.1-1, FUNCTION 1.b and note (c)
STS Table 3.3.2.1-1, FUNCTION 2 and note (f)	

In all of the above cases the BPWS and RWM/RPC are applicable in MODES 1 and 2 when power is $\leq$ 10% RTP (i.e., below the LPSP), for both reactor startup and shutdown.

For completeness, the above STS are provided in Appendix A.

This LTR documents an acceptable alternate approach for complying with the BPWS. After this LTR is NRC approved, it is expected that plant-specific TS BASES will be updated to reference this LTR and incorporate the operating recommendations herein, for using this alternate BPWS approach during the reactor shutdown process. With the TS Bases appropriately updated, most of the TS locations (listed above) do not need to be changed. The STS locations that are subjected to change are provided in Section 6.

4. SAFETY AND TECHNICAL EVALUATIONS

The BPWS was originally focused on application to reactor startups; however, it was also applied to reactor shutdowns, because of the potential for high worth rod patterns during the shutdown process. However, confirming that control rods are coupled prior to decreasing power below the LPSP eliminates the potential for a CRDA during the reactor shutdown process, and thus, the need for banking. This section addresses steps to ensure control rod coupling integrity for the control rods not fully inserted prior to reaching the LPSP, which will then permit control rods to be fully inserted in a single step, when the reactor is below the LPSP.

The function of the banking steps of the BPWS is to minimize the potential reactivity increase from a postulated CRDA at low power levels. Therefore, if the possibility for a control rod to drop can be eliminated, then the banking steps at low power levels are not needed to ensure the applicable event limits cannot be exceeded. It is not possible to drop a control rod that is coupled to or in contact with its CRD, and thus, if the controls specified herein are applied, a CRDA is not a credible event for this situation while inserting control rods during the reactor shutdown process. The following discusses how control rod coupling is confirmed prior to reaching the STS BPWS applicability limit during the reactor shutdown process, thereby eliminating the need for the control rod banking steps.

The STS from NUREG 1433 and NUREG 1434 require coupling checks be performed any time a control rod is fully withdrawn. Coupling is confirmed by a continuous indication of position "48" on the control rod position indication display while the operator attempts to withdraw the control rod past position 48. If the control rod is not coupled, the position 48 indication will extinguish, the over travel light will light, and an alarm sounds. Based on STS, the following statements are deduced:

- If a rod has been fully withdrawn during the cycle and then determined to be coupled, and the rod has not been moved from position 48, then coupling integrity is assured, because of the improbability of a control rod becoming decoupled when it has not been moved.
- If after a coupling check is performed for a control rod, the rod is inserted and then withdrawn to the full out position, it again requires a coupling check. However, if the rod is withdrawn to an intermediate position, coupling integrity is not assured for this rod.
- If a rod has been checked for coupling at notch position 48 and the rod has since only been moved inward, no subsequent coupling check is required, because control rod insertion maintains contact between the control rod and the drive.

To ensure that control rods are not stuck and are not decoupled, the surveillances within STS 3.1.3 (Control Rod OPERABILITY) require stuck rod and coupling checks to be routinely performed. For stuck rod checks, the fully withdrawn rods are usually inserted one notch and withdrawn one notch. For a coupling check, an operator typically attempts to withdraw the control rod past notch position 48, when the rod position is indicated at notch position 48. If no over travel indication is observed, then the coupling check is satisfactory. The routine CRD coupling checks ensure control rod coupling integrity for the fully withdrawn rods, and are typically performed every seven days.

After startup, 80 to 90% of control rods would have been checked for coupling, because they would be fully withdrawn during power operation. The remaining control rods would be checked at some time during the cycle as control rods are alternated in and out of the core. For an end of cycle shutdown, all rods are typically fully withdrawn, and therefore, checked for coupling. To eliminate the possibility of a CRDA, the proposed controls require that any partially inserted control rods, which have not been confirmed to be coupled since their last withdrawal, be fully inserted prior to reaching the LPSP.

However, if a rod has been checked for coupling at notch position 48 and the rod has since only been moved inward, this rod is in contact with its drive and thus is not required to be fully inserted prior to reaching the LPSP. However, if only inward movement cannot be confirmed for a partially inserted control rod, the control rod shall be fully inserted prior to reaching the LPSP.

Based on the discussion above, it is concluded that partially inserted rods that are not assured to be in contact with their drives would be required to be fully inserted before the power is reduced to the LPSP. The remaining rods are not susceptible to a CRDA, making the banking steps during the reactor shutdown process below the LPSP unnecessary.

If a plant is required to be shutdown and all rods not confirmed of coupling cannot be fully inserted prior to the power reaching the LPSP (e.g., shortly after a startup), then the proposed changes to the shutdown process may not be implemented. However, after all rods that are not confirmed of coupling are fully inserted, the proposed shutdown process is allowed. When there is a withdrawn rod that is not confirmed to be coupled, the standard (e.g., Reference 2) BPWS steps must be followed below the LPSP or a scram is required to protect against the CRDA.

Additionally, if a plant is in the process of shutting down while using the improved BPWS control rod insertion process below the LPSP, no control rod shall be withdrawn unless the control rod pattern is in compliance with the standard BPWS requirements (e.g., at about 75% or higher control rod density). This assures that rod withdrawals comply with standard BPWS withdrawal requirements.

To be allowed to continue operating with a stuck withdrawn or partially inserted control rod, the CRD must be inserted as much as possible and then disarmed, an evaluation of adequate (per TS requirements) cold shutdown margin (SDM) is required, and an evaluation that justifies (consistent with STS 3.1.3) operating with a stuck rod has been approved. The SDM must be evaluated (by measurement or analysis) with the stuck control rod at its stuck position and the highest worth OPERABLE control rod assumed to be fully withdrawn. The SDM evaluation demonstrates adequate SDM and that MODE 4 can be obtained. Inserting the CRD as much as possible and disarming it assures that no SOE can cause the stuck rod to drop, and the stuck rod can then be considered as coupled. In this case, both SDM and CRDA concerns are alleviated, and thus, use of the improved BPWS control rod insertion process does not affect plant safety and is permitted.

5. PLANT IMPLEMENTATION

To implement the proposed change to the shutdown process, the following guidance should be reflected in plant procedures.

A. Actions Prior to Reducing Power to the LPSP

Fully Withdrawn Control Rods

Before reducing power below the LPSP, operations shall confirm control rod coupling integrity for all rods that are fully withdrawn. (If rod coupling has been checked twice or has been verified, and the rod has not been subsequently inserted and withdrawn, the coupling check need not be repeated prior to reducing power below the LPSP.)

Note: The coupling confirmation check is unchanged. This check is performed by withdrawing the CRD to position "48" (full-out) and attempting to withdraw the control rod past position 48. Coupling is confirmed by a continuous indication of "48" on the rod position indication display. An over travel would indicate the CRD has traveled beyond the full-out position which is indicative of a decoupled control rod. Existence of an over travel condition is by: (1) position 48 indication extinguished, (2) lighting of the over travel light: and (3) sounding of the over travel alarm.

A rod coupling is considered confirmed when there have been two documented coupling checks or one verified and documented coupling check. (This step ensures that no SOE can result in an incorrect coupling check.)

Control Rods In Intermediate Positions

Control rods that have not been confirmed coupled (at notch position 48 since they were last withdrawn) must be fully inserted prior to power reduction to the LPSP. However, if a rod has been checked for coupling at position 48 and the rod has since only been moved inward, this rod does not need to be inserted prior to reaching the LPSP.

Fully Inserted Control Rods

No action is required.

After power is reduced to the LPSP and all rods that were not confirmed coupled have been fully inserted, the RWM/RPC may be bypassed (if needed).

If shutdown is required and all rods, which are not confirmed coupled, cannot be fully inserted prior to the power dropping below the LPSP (such as shortly after a startup), then the standard (e.g., Reference 2) BPWS must be observed below the LPSP or a scram is required. However, during the shutdown process using the standard BPWS and after all rods, which were not confirmed coupled, have been fully inserted, the improved BPWS control rod insertion process may be used.

B. Actions Below the LPSP

As much as reasonably possible, the control rod groups should be inserted in the same order as specified for the standard BPWS. (This is considered a matter of good practice, because it allows for a faster restart, if the reactor shutdown is aborted.) All the control rods in a group should be fully inserted prior to inserting rods in the next group.

The rods may be inserted from notch position 48 to notch position 00 without stopping at intermediate positions.

Note: This sequence may be programmed into the RWM/PRCS/RSCS, if a plant's design provides this capability.

C. Control Rod Withdrawal Below LPSP

When a plant is in the process of shutting down while fully inserting control rods in a single step below the LPSP, no control rod shall be withdrawn unless the control rod pattern is in compliance with standard BPWS requirements.

D. Inoperable and Stuck Control Rods

If a plant has only one stuck control rod with its drive inserted as much as possible and disarmed, and continuous operation has been allowed per STS 3.1.3, then use of the improved BPWS control rod insertion process is allowed. In all other cases with stuck control rods, the improved BPWS control rod insertion process is not applicable, and the current requirements for inoperable and stuck rods shall be followed.

6. PROTOTYPICAL TECHNICAL SPECIFICATIONS CHANGES

6.1 NUREG 1433 BWR/4 STS Change

If needed on a plant-specific basis, qualify note (f) of Table 3.3.2.1-1 by adding "*, except during the reactor shutdown process if the coupling of each withdrawn control rod has been confirmed*" to the end of the note.

(f) With THERMAL POWER ≤ [10]% RTP[*, except during the reactor shutdown process if the coupling of each withdrawn control rod has been confirmed*].

It is envisioned that the above change "if needed" would be necessary only for those plants that do not have the ability to readily modify or reprogram their RWM. If a plant is not able to revise their RWM, the above TS change would allow the RWM to be bypassed, and thus, the shutdown sequence described herein could be utilized. For most, if not all, plants with TS based upon NUREG 1433, the above change to their plant-specific TS would not be warranted.

6.2 NUREG 1434 BWR/6 STS Change

If needed on a plant-specific basis, qualify note (c) of Table 3.3.2.1-1 by adding "*, except during the reactor shutdown process if the coupling of each withdrawn control rod has been confirmed*" to the end of the note.

(c) With THERMAL POWER ≤ [10]% RTP[*, except during the reactor shutdown process if the coupling of each withdrawn control rod has been confirmed*].

It is envisioned that the above change "if needed" would be necessary for most of the BWR/6 plants, because they do not have the ability to readily modify or reprogram their RPC. If a plant is not able to revise their RPC, the above TS change would allow the RPC to be bypassed, and thus, the shutdown sequence described herein could be utilized. For most plants with TS based upon NUREG 1434, the above change to their plant-specific TS would be warranted. Following submittal of this LTR to the NRC, it is envisioned that a Technical Specification Task Force (TSTF) submittal will be generated to capture the above change to NUREG 1434.

7. SAFETY AND PLANT BENEFITS

The following section discusses benefits for the elimination of control rod banking during the reactor shutdown process. Aspects addressed include reactivity management, human factors, scram avoidance and equipment duty.

7.1 Reactivity Management

By eliminating the banking steps during the reactor shutdown process, negative reactivity can be more rapidly inserted into the core. Unlike startup in which positive reactivity insertions must be slow and controlled, it is acceptable to rapidly insert negative reactivity while shutting down.

A faster reactor shutdown achieves the All Rods In condition prior to significant reactor cool down. Because core reactivity normally increases with decreasing reactor coolant temperature, achieving All Rods In faster reduces the potential for re-criticality during the control rod insertion process. That is, if the negative reactivity insertion rate due to control rod movements is more than the positive reactivity insertion rate due to cool down, then a re-criticality cannot occur.

7.2 Human Performance

Eliminating banking during reactor shutdown decreases the number of steps from about 400 to 150 for a medium size reactor. This reduces the number of potential reactivity control errors that could occur, because it reduces the number of operator actions below the LPSP to achieve reactor shutdown.

7.3 Scram Avoidance

The ability to achieve a faster shutdown by fully inserting control rods in a single step helps eliminate the need to manually scram the reactor. Using the improved BPWS control rod insertion process reduces the potential for improperly entering into a control rod pattern in which rods cannot be moved, and thus, requiring a scram.

7.4 Equipment Duty

The reduction in the number of control rod positioning steps prevents unnecessary control rod manipulations. This reduces the duty on the Reactor Manual Control System and CRD hardware, which improves equipment reliability because it reduces the number of operations to achieve reactor shutdown. In addition, avoiding scrams results in less duty on the CRD system components, and thus, also improves CRD component reliability.

8. REFERENCES

1. GE Nuclear Energy, "GESTAR II General Electric Standard Application for Reactor Fuel," US Supplement NEDE-24011-P-A-14-US, Class III (Proprietary), June 2000.
2. General Electric Co., Licensing Topical Report, "Banked Position Withdrawal Sequence," NEDO-21231, Class I (non-proprietary), January 1977.
3. USNRC, "Standard Technical Specifications General Electric Plants, BWR/4," NUREG-1433, Rev. 2, June 2001.
4. USNRC, "Standard Technical Specifications General Electric Plants, BWR/6," NUREG-1434, Rev. 2, June 2001.

APPENDIX A

APPLICABLE STANDARD TECHNICAL SPECIFICATIONS

For completeness, the current Standard Technical Specifications (STS) from NUREG 1433 (Reference 3) and NUREG 1434 (Reference 4), which address the subjects discussed in this report, are provided below.

A.1 NUREG 1433 BWR/4 STS

STS 3.1.3; CONDITION D and REQUIRED ACTION D.1:

D. --------------------------- - NOTE - Not applicable when THERMAL POWER > [10]% RTP. --------------------------- Two or more inoperable control rods not in compliance with banked position withdrawal sequence (BPWS) and not separated by two or more OPERABLE control rods.	D.1 Restore compliance with BPWS. OR D.2 Restore control rod to OPERABLE status.

STS 3.1.6; LCO 3.1.6:

OPERABLE control rods shall comply with the requirements of the [banked position withdrawal sequence (BPWS)].

STS 3.1.6; CONDITIONS A and B, and REQUIRED ACTIONS A.1 and B.1:

CONDITION	REQUIRED ACTION
A. One or more OPERABLE control rods not in compliance with [BPWS].	A.1 --------------------------------- - NOTE - Rod worth minimizer (RWM) may be bypassed as allowed by LCO 3.3.2.1, "Control Rod Block Instrumentation." --------------------------------- Move associated control rod(s) to correct position.

B. Nine or more OPERABLE control rods not in compliance with [BPWS].	B.1 --- **- NOTE -** Rod worth minimizer (RWM) may be bypassed as allowed by LCO 3.3.2.1. --- Suspend withdrawal of control rods.

STS SR 3.1.6.1:

SR 3.1.6.1 Verify all OPERABLE control rods comply with [BPWS].

STS 3.3.2.1, CONDITION C:

C. Rod worth minimizer (RWM) inoperable during reactor startup

STS 3.3.2.1, REQUIRED ACTION C.2.2:

C.2.2 Verify movement of control rods is in compliance with banked position withdrawal sequence (BPWS) by a second licensed operator or other qualified member of the technical staff.

STS 3.3.2.1, CONDITION D and REQUIRED ACTION D.1:

D. RWM inoperable during Reactor shutdown	D.1 Verify movement of control rods is in compliance with BPWS by a second licensed operator or other qualified member of the technical staff.

STS SR 3.3.2.1.8:

SR 3.3.2.1.8 Verify control rod sequences input to the RWM are in conformance with BPWS.

STS Table 3.3.2.1-1, FUNCTION 2 and note (f):

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
2. Rod Worth Minimizer	1(f), 2(f)	[1]	SR 3.3.2.1.2 SR 3.3.2.1.3 SR 3.3.2.1.5 SR 3.3.2.1.8	NA

(f) With THERMAL POWER ≤ [10]% RTP.

A.2 NUREG 1434 BWR/6 STS

STS 3.1.3; CONDITION D and REQUIRED ACTION D.1:

(Same as for NUREG 1433.)

STS 3.1.6; LCO 3.1.6:

(Same as for NUREG 1433.)

STS 3.1.6; CONDITION A:

(Same as for NUREG 1433.)

STS 3.1.6; CONDITION B:

(Same as for NUREG 1433.)

STS SR 3.1.6.1:

(Same as for NUREG 1433.)

STS Table 3.3.2.1-1, FUNCTION 1.a and note (c):

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	SURVEILLANCE REQUIREMENTS
1.b. Rod pattern controller	1(c), 2(c)	[1]	SR 3.3.2.1.3 SR 3.3.2.1.4 SR 3.3.2.1.5 SR 3.3.2.1.7 SR 3.3.2.1.9

(c) With THERMAL POWER ≤ [10]% RTP.

APPENDIX B

NRC REQUESTS FOR ADDITIONAL INFORMATION AND ACCEPTED RESPONSES

GE Nuclear Energy

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MFN 04-047
April 21, 2004

U.S Nuclear Regulatory Commission
Document Control Desk
Washington, D.C. 20852-2738

Attention: Chief, Information Management Branch
Program Management
Policy Development and Analysis Staff

Subject: **Responses to Informal NRC RAIs and Concerns From A 03/23/04 Conference Call Regarding The Improved BPWS, LTR NEDO-33091**

During the NRC Staff review of the Reference 1 Licensing Topical Report (LTR), through emails and a 03/23/04 GE-NRC conference call the NRC provided a number of informal requests for additional information (RAIs) and voiced some other concerns with respect to Reference 1. Enclosure 1 provides GE's responses to all of those RAIs and concerns.

If you have any questions, please contact, Kurt Schaefer at (408) 925-2426 or myself.

Sincerely,

for

George Stramback
Manager, Regulatory Services
GE Nuclear Energy
(408) 925-1913
george.stramback@gene.ge.com

Project No. 710

Reference:

1. BWR Owners' Group Licensing Topical Report, *Improved BPWS Control Rod Insertion Process*, NEDO-33091 (non-proprietary), April 2003.

Enclosures:

1. Responses to NRC Staff Requests For Additional Information on NEDO-33091.

cc: AB Wang (NRC)
Bo Pham (NRC)
J. Tuttle (GNF)
DRF 0000-0026-2578

Responses to NRC Staff Requests For Additional Information On NEDO-33091

1. The LTR states that it is impossible for a control rod to become decoupled during the shutdown process following the improved BPWS. Please provide detailed information about the different types of control rod drives (CRDs) used for BWR-2 to BWR-6 plants with respect to the coupling between the CRD and control rod during shut-down. Based on this information, please also explain why it is not possible for the CRD and control rod to be decoupled.

   Response:
   All Reg. Guide 1.70, Rev.3 BWR UFSARs contain failure mode and effects analyses (FMEA) for the CRD system. Attachment 1 includes some pages from the FMEA in a typical BWR UFSAR (i.e., Limerick). As stated in the last paragraph of page 4.6-18, no single failure can of itself initiate a rod withdrawal. This statement is even more applicable, when the collet fingers are locked, such as when there is no CR movement or the CR is being inserted. (See Limerick Figure 4.6-3.) For a CR to withdraw, the six collet fingers must be unlocked so that the index tube can be lowered without being latched to a stop by the collet fingers. During CR insertion, if a mechanical/hydraulic failure occurs that stops all upward hydraulic force, the most the CR could withdraw is 6 inches to the next index tube latching indentation.

   As summarized in UFSAR subsection 4.6.2.3.1.4, a limiting multi-failure scenario with the collet fingers remaining open has been evaluated. The resulting maximum CR withdrawal speed is 2 ft/sec, which is bounded by the 5 ft/sec CR drop speed assumed in the Control Rod Drop Accident (CRDA) analyzed in UFSAR Chapter 15, as discussed in UFSAR subsection 4.6.2.3.2.2.

   Attachment 1 also provides the UFSAR Control Rod Drive (CRD) and Control Rod (CR) design figures for a BWR/2 (Oyster Creek), BWR/3 (Monticello), BWR/4 (Limerick), BWR/5 (La Salle) and BWR/6 (Perry). For all BWR/2-6 plants, these figures demonstrate that a CR sits atop its associated CRD, and that it is physically impossible for a CR to move down past (i.e., to be withdrawn) its CRD.

   Normally, a CR cannot become uncoupled from its CRD, because the CRD spud ("6-fingers") mechanically locks into the CR socket, and operator action is required to unlock this coupling. It is possible, due to a CRD assembly error, for a CRD to not properly lock into CR. In this case, CR movement (withdrawal and insertion) and scram are not impaired, because gravitational force will maintain the CR on top of its CRD. Plant Technical Specifications already provide provisions for coupling checks and steps to be taken if a CR is uncoupled from its drive.

   The improved BPWS only involves CR insertion steps that prevent the CRs from dropping (i.e., separating from CRD), and does not involve/affect any of the functions discussed above.

2. In the GE Proprietary Information Class III document (WE-1048), which describes the control rod drop design basis accident, page 6-16, Section 6.4, Control Rod Drop Accident, states the following:

   "One of the four design basis accidents is a control rod drop. The design basis control rod drop accident is defined as the complete (but unnecessarily sudden) rupture, breakage, or disconnection of a fully inserted control rod drive from its cruciform control blade at or near the coupling and in such a way that the blade somehow becomes stuck at its location inserted."

   This description includes rupture and breakage as mechanisms that may trigger the rod drop accident (RDA). Please explain why the BWROG/GE changed from this definition to the exclusion of CRD failure as one of the causes for RDA. Is the BWROG/GE proposing a new position that CRD failure will not be considered as the cause for RDA in future analyses?

   Response:
   NEDO-33091 Rev. 2.0, "BWR Owners' Group Licensing Topical Report: Improved BPWS Control Rod Insertion Process", describes an improved control rod (CR) insertion process. The rod drop accident scenario, also described in the report (1.0 Background), requires the following conditions:

   1. The CR is not coupled to the control rod drive (CRD).
   2. The CR is stuck.
   3. The CRD is withdrawn, leaving the stuck CR suspended in position.
   4. The CR subsequently becomes unstuck and drops freely at 3.11 feet/second.

   Because the Improved BPWS process, described by NEDO-33091, is only for CR insertion and does not involve CR withdraw, then one of the primary factors in the rod drop accident (item 3, above) is eliminated. Therefore, the rod drop accident scenario is eliminated from consideration for this CR insertion process.

   Additionally, there are no single failure scenarios that will result in a CR withdrawal/ejection/drop, including CRD withdrawals with CR velocities corresponding to the control rod drop accident. "There are no known single malfunctions that cause the unplanned (i.e., no operator action is involved) withdrawal of even a single control rod." [See (in Attachment 2) Limerick UFSAR 4.6.2.3.2.2 for BWR/2-5 CRDs and River Bend UFSAR 4.6.2.3.2.2 for BWR/6 CRDs.] This applies to both BWR 2-5 CRDs and BWR/6 CRDs, because they are essentially same regarding the design of the locking mechanism and the mechanism for CRD insertion, withrawal and scram. (The primary difference between these CRDs is the mechanism for slowing the control rod at the end of a scram stroke.) The scenarios, which result in CR withdrawal, require multiple failures combined with a CR withdrawal signal (not part of the improved CR insertion process). Otherwise, the CRD latching mechanism, the hydraulic conditions and/or the control rod drive housing support prevent CR withdrawal. [See (in Attachment 2) Limerick UFSAR 4.6.2.3.2.2.1 through 4.6.2.3.2.2.12 and River Bend UFSAR 4.6.2.3.2.2.1 through 4.6.2.3.2.2.12 for BWR/6 CRDs.]

For example, if the drive housing fails at the attachment weld (River Bend UFSAR 4.6.2.3.2.2.1), "the CRD and housing would be blown downward against the support structure" and "if the collet were to remain latched" (which is expected in this scenario), "no further control rod ejection would occur." Also, the "maximum deflection is approximately 3 in." If the "failure were to occur while the control rod is being withdrawn" (i.e., during a planned withdrawal) and "if the collet were to stay unlatched", then "the steady-state rod withdrawal velocity would be 0.3 ft/sec" and would continue "until driving pressure was removed from the pressure-over port" (i.e., the single notch out sequence stops automatically or the operator terminates the continuous withdraw command). Note that unlike the latter scenario, the improved BPWS CR insertion process does not initiate a CR withdrawal command.

Regardless of the scenario, all CRD failure modes and effects evaluations in the UFSARs are unaffected by the Improved BPWS, which only involves CR insertion changes.

3. The improved BPWS will significantly reduce the time of the normal shut down process. The LTR states that the total rod manipulation steps for shutting down a medium sized reactor will be reduced from approximately 400 to 150 steps. While this reduction allows for faster insertion of negative reactivity, the rapid power decrease may cause quenching of the reactor pressure vessel or the necessary speeding up of other processes. Please provide a discussion about the impact of the improved BPWS on other procedures in this regard, if any. In particular, please address how the reactor vessel Pressure-Temperature limit will be maintained during the shut down process.

   Response:
   The improved BPWS does not adversely (if at all) affect any normal shutdown process, including maintaining the reactor vessel cool down time rate within its Pressure-Temperature limits. Current operating procedures already account for the effects of the most limiting (fastest negative reactivity) CR insertion scenario, i.e., scram. Operations control the cool down process by controlling reactor dome pressure, coolant flow and coolant temperature, regardless of the CR insertion process.

4. Using the improved BPWS process, control rods not confirmed to be coupled are required to be fully inserted prior to reducing power below the low power set point (LPSP). How does the improved BPWS ensure this full insertion prior to the LPSP? Is it possible for some of the un-confirmed rods to not be fully inserted with power dropping below the LPSP?

   Response:
   As with the original BPWS, the insertion steps are controlled by plant procedures, and thus, to implement the improved BPWS, plant procedures will be updated. Section 5, Item A of the LTR provides guidance to be used in updating plant procedures, to ensure that any un-confirmed CR are fully inserted prior to reaching the LPSP. The probability of an un-confirmed rod not being fully inserted, while the power drops below the LPSP, is remote because of the followings:

   a. The LTR requires that each CR that has not been confirmed coupled (since its last withdrawal) to be fully inserted prior to reducing power below the LPSP. Therefore, the operator is aware of the status of each rod.

   b. The control rods are inserted, as much as reasonably possible, in the same order as specified for the standard BPWS. Therefore, the operator is absolutely aware of the status of each control rod before the shutdown process begins.

   c. The LTR states that operations shall confirm control rod coupling integrity for all rods that are fully withdrawn and there would be two documented coupling checks or one verified and documented coupling check. Therefore, the probability of not confirming the coupling integrity of the fully withdrawn rod is remote.

   For normal shutdowns, it is expected that all potentially un-confirmed CRs will be inserted prior to reducing power to 40%, which is significantly above the LPSP. However, the last paragraph in LTR page 5-1 addresses the postulated scenario of a plant being below LPSP prior to having all un-confirmed CRs fully inserted. The LTR states, "*If shutdown is required and all rods, which are not confirmed coupled, cannot be fully inserted prior to the power dropping below the LPSP (such as shortly after a startup), then the standard (e.g., Reference 2) BPWS must be observed below the LPSP or a scram is required. However, during the shutdown process using the standard BPWS and after all rods, which were not confirmed coupled, have been fully inserted, the improved BPWS control rod insertion process may be used.*"

   Therefore, the improved BPWS is only allowed to be used when all un-confirmed CRs are fully inserted prior to power dropping below the LPSP.

5. Page 2-1 of the LTR states, "require control rod confirmations. . . ." Does this requirement apply to all control rods? If not, please clarify which control rods need to be confirmed.

   Response:
   Currently, plant Technical Specifications require all CRs to be coupled to their CRDs. Technical Specifications Surveillances require coupling checks (a) prior to reactor criticality after completing core alterations, (b) anytime a CR is withdrawn to the full out position, and (3) following maintenance on or modification to a CR or CRD system. These requirements are not affected by the improved BPWS.

   LTR Section 5 Item A discusses how the improved BPWS goes beyond the above requirements. To avoid a single operator error (SOE), rod couplings must be checked twice or be verified by a second operator. This applies to all CRs that are not already fully inserted.

6. Will the improved BPWS replace all existing BPWSs in U.S. BWRs? Will there be any exceptions?

Response:
The improved BPWS does not replace the existing BPWS. The improved BPWS only provides an alternate approach to the existing BPWS with respect to control rod insertions, when a plant is being shutdown and all non-fully inserted control rods have been confirmed to be coupled. The improved BPWS may be used by all U.S. BWRs, but it is not mandatory. The implementation of the improved BPWS is a plant-specific decision.

Responses to NRC Staff Requests For Additional Information On NEDO-33091

Addressing NRC Concerns From the 3/23/05 Conference Call & Clarification of the Response to Improved BPWS LTR RAI 1.1

1. Explanation of the difference between the velocities stated in UFSAR 4.6.2.3.1.4 (2 fps) and 4.6.2.3.2.2.7 (11.8 fps).

The velocity stated in 4.6.2.3.1.4 (2 fps) is the result of the scenario described in 4.6.2.3.2.2.2.1 Pressure-Under (Insert) Line Break. In this scenario, failure of the insert line occurs while the rod is being withdrawn. The flange ball check valve would lift and close the flow path out through the failed insert line due to the differential pressure. Then the under-piston water (while CRD is withdrawing) would be routed to the flange vessel ports (under the ball check valve) and discharged into the reactor vessel. Because the hydraulic resistance of this flow path is less than the normal resistance through the Directional Control Valve (i.e., a throttled valve), the velocity increases to 2 fps if the collet is stuck open (in the unlatched position). Normally, at this velocity (2 fps) the hydraulic force would not be sufficient to hold the collet open and the collet would be forced into the latched position, hence, stopping rod withdrawal. In summary, the operator selects this rod for withdrawal and two failures are assumed: failed insert line and collet failure to latch during the higher than normal velocity condition.

The velocity stated in 4.6.2.3.2.2.7 (11.8 fps) is the result of the failure of the flange ball check valve plug while the rod is being withdrawn. The under-piston water (while CRD is withdrawing) would be discharged through the hole in the side of the flange and directly out to atmosphere. Because the hydraulic resistance of this flow path is significantly less than the normal resistance through the Directional Control Valve (i.e., a throttled valve) or through the flange vessel ports (mentioned above), the velocity increases to 11+ fps if the collet is stuck open. Normally, at this high velocity the hydraulic force would not be sufficient to hold the collet open and the collet would be forced into the latched position, hence, stopping rod withdrawal. In summary, the operator selects this rod for withdrawal and two failures are assumed: failed flange ball check valve plug and collet failure to latch during the higher than normal velocity condition.

2. Explanation of the failure scenario described in 4.6.2.3.2.2.11 relative to single failure.

Two cases could lead to continuous rod withdrawal following a deliberate withdrawal command.

Case 1: Following a withdrawal command and assuming the signal terminates as expected, a failure of the Directional Control Valve to close would cause the rod to continue to withdraw.

Case 2: Following a withdrawal command and assuming the signal terminates as expected, a failure of the collet to return to its latched position would cause the rod to continue to drift out.

Yes, a single failure (failure of Directional Control Valve to close or failure of the collet to return to its latched position) would cause the rod to continuously drift out. In summary, an inadvertent continuous rod withdrawal must be first initiated by the operator's withdrawal command followed by one of these failures.

3. Explanation of the statement "unplanned withdrawal" in 4.6.2.3.2.2.

In the 4.6.2.3.2.2 statement "There are no known single malfunctions that cause the unplanned withdrawal of even a single control rod," an "unplanned withdrawal" is a CR withdrawal that does not involve any operator action, e.g., the operator does not initiate a CR withdrawal.

RAI 1.1 has been renumbered as RAI 2, above. The response to RAI 2, above, is the clarified response to RAI 1.1.

Responses to NRC Staff Requests For Additional Information On NEDO-33091

Attachment 1

UFSAR CRD FMEA and CRD & CR UFSAR Design Figures for BWR/2 - 6

As is illustrated in Figure 4.6-3, the locking mechanism is a ratchet-type device and does not interfere with rod insertion. The speed at which the drive moves is determined by the flow through the insert speed control valve, which is set for approximately 4 gpm for a shim speed (nonscram operation) of 3 in/sec. During normal insertion, the pressure on the downstream side of the speed control valve is 90-100 psi above reactor vessel pressure. However, if the drive slows for any reason, the flow through and the pressure drop across the insert speed control valve decreases; the full differential pressure (260 psi) is then available to cause continued insertion. With 260 psi differential pressure acting on the drive piston, the piston exerts an upward force of 1040 pounds.

4.6.1.2.5.2 Rod Withdrawal

Rod withdrawal is, by design, more involved than insertion. The collet fingers (latch) must be raised to reach the unlocked position (Figure 4.6-3). The notches in the index tube and the collet fingers are shaped so that the downward force on the index tube holds the collet fingers in place. The index tube must be lifted before the collet fingers can be released. This is done by opening the drive insert valves (in the manner described in the preceding paragraph) for approximately one second. The withdraw valves are then opened, applying driving pressure above the drive piston, and opening the area below the piston to the exhaust header. Pressure is simultaneously applied to the collet piston. As the piston raises, the collet fingers are cammed outward, away from the index tube, by the guide cap.

The pressure required to release the latch is set and maintained at a level high enough to overcome the force of the latch return spring, plus the force of reactor pressure opposing movement of the collet piston. When this occurs, the index tube is unlatched and free to move in the withdraw direction. Water displaced by the drive piston flows out through the withdraw speed control valve, which is set to give a rod shim speed of approxiately 3 in/sec. The entire valving sequence is automatically controlled, and is initiated by a single operation of the rod withdraw switch.

Rod withdrawal will not occur without permissive operator action. Following a deliberate operator withdrawal action, a rod drift could occur due to failure of its collet assembly to return to the locked position. The operator can interrupt this withdrawal with a scram or an insert signal. No single failure can of itself initiate a rod withdrawal.

4.6.2.3.1 Control Rods

4.6.2.3.1.1 Materials Adequacy Throughout Design Lifetime

The adequacy of the materials throughout the design life is evaluated in the mechanical design of the control rods. The primary materials, boron carbide (B_4C) powder, solid hafnium, and 304 austenitic stainless steel, have been found suitable in meeting the demands of the BWR environment.

4.6.2.3.1.2 Dimensional and Tolerance Analysis

Layout studies are done to ensure that, given the worst combination of part tolerance ranges at assembly, no interference exists that will restrict the passage of control rods. In addition, preoperational verification is made on each control blade system to show that the acceptable levels of operational performance are met.

4.6.2.3.1.3 Thermal Analysis of the Tendency to Warp

The various parts of the control rod assembly remain at approximately the same temperature during reactor operation, negating the problem of distortion or warpage. What little differential thermal growth that could exist is allowed for in the mechanical design. A minimum axial gap is maintained between absorber rod tubes and the control rod frame assembly for this purpose. Use of dissimilar metals (stainless steel and hafnium) is evaluated to ensure that any effects due to thermal expansion or irradiation growth are acceptable.

4.6.2.3.1.4 Forces for Expulsion

An analysis has been performed which evaluates the maximum pressure forces which could tend to eject a control rod from the core. The results of this analysis are given in Section 4.6.2.3.2.2.2. In summary, if the collet were to remain open, which is unlikely, calculations indicate that the steady-state control rod withdrawal velocity would be 2 ft/sec for a pressure-under line break, the limiting case for rod withdrawal.

4.6.2.3.1.5 Functional Failure of Critical Components

The consequences of a functional failure of critical components have been evaluated and the results are covered in Section 4.6.2.3.2.2.

4.6.2.3.1.6 Precluding Excessive Rates of Reactivity Addition

In order to preclude excessive rates of reactivity addition, analysis has been performed both on the velocity limiter device, and the effect of probable control rod failures (Section 4.6.2.3.2.2).

4.6.2.3.2 Control Rod Drives

4.6.2.3.2.1 Evaluation of Scram Time

The rod scram function of the CRD system provides the negative reactivity insertion required by safety design basis as stated in Section 4.6.1.1.1. The scram time shown in the description is adequate as shown by the transient analyses of Chapter 15.

4.6.2.3.2.2 Analysis of Malfunction Relating to Rod Withdrawal

There are no known single malfunctions that cause the unplanned withdrawal of even a single control rod. However, if multiple malfunctions are postulated, studies show that an unplanned rod withdrawal can occur at withdrawal speeds that vary with the combination of malfunctions postulated. In all cases the subsequent withdrawal speeds are less than that assumed in the rod-drop accident analysis as discussed in Chapter 15. Therefore, the physical and radiological consequences of such rod withdrawals are less than those analyzed in the rod-drop accident.

4.6.2.3.2.2.1 Drive Housing Failure at Attachment Weld

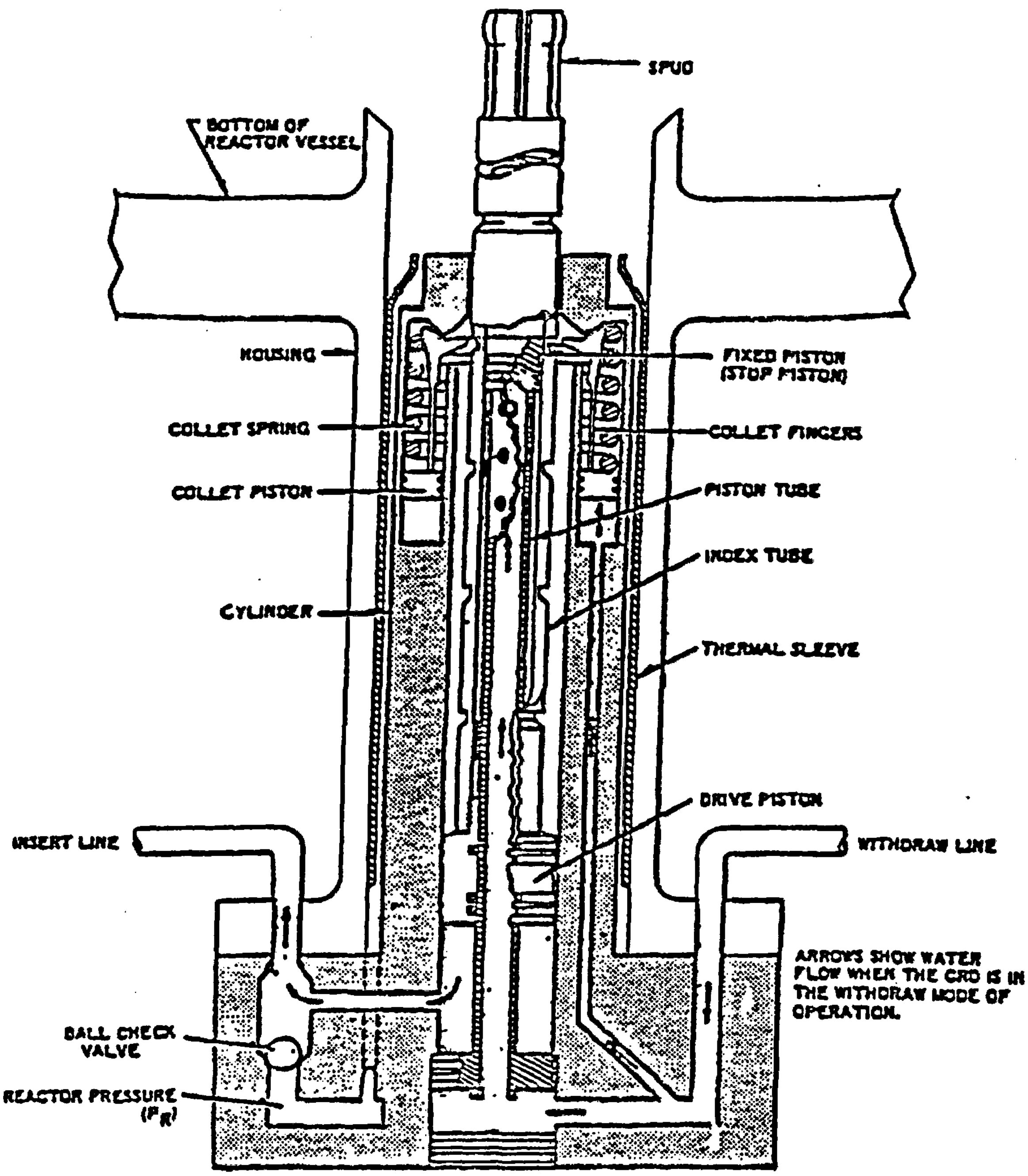
The bottom head of the reactor vessel has a penetration for each CRD location. A drive housing is raised into position inside each penetration and is fastened by welding. The drive is raised into the drive housing and bolted to a flange at the bottom of the housing. The housing material is seamless, Type 304 stainless steel pipe with a minimum tensile strength of 75,000 psi. The basic failure considered here is a complete circumferential crack through the housing wall at an elevation just below the J-weld.

Static loads on the housing wall include the weight of the drive and the control rod, the weight of the housing below the J-weld, and the reactor pressure acting on the 6 inch diameter cross-sectional area of the housing and the drive. Dynamic loading results from the reaction force during drive operation.

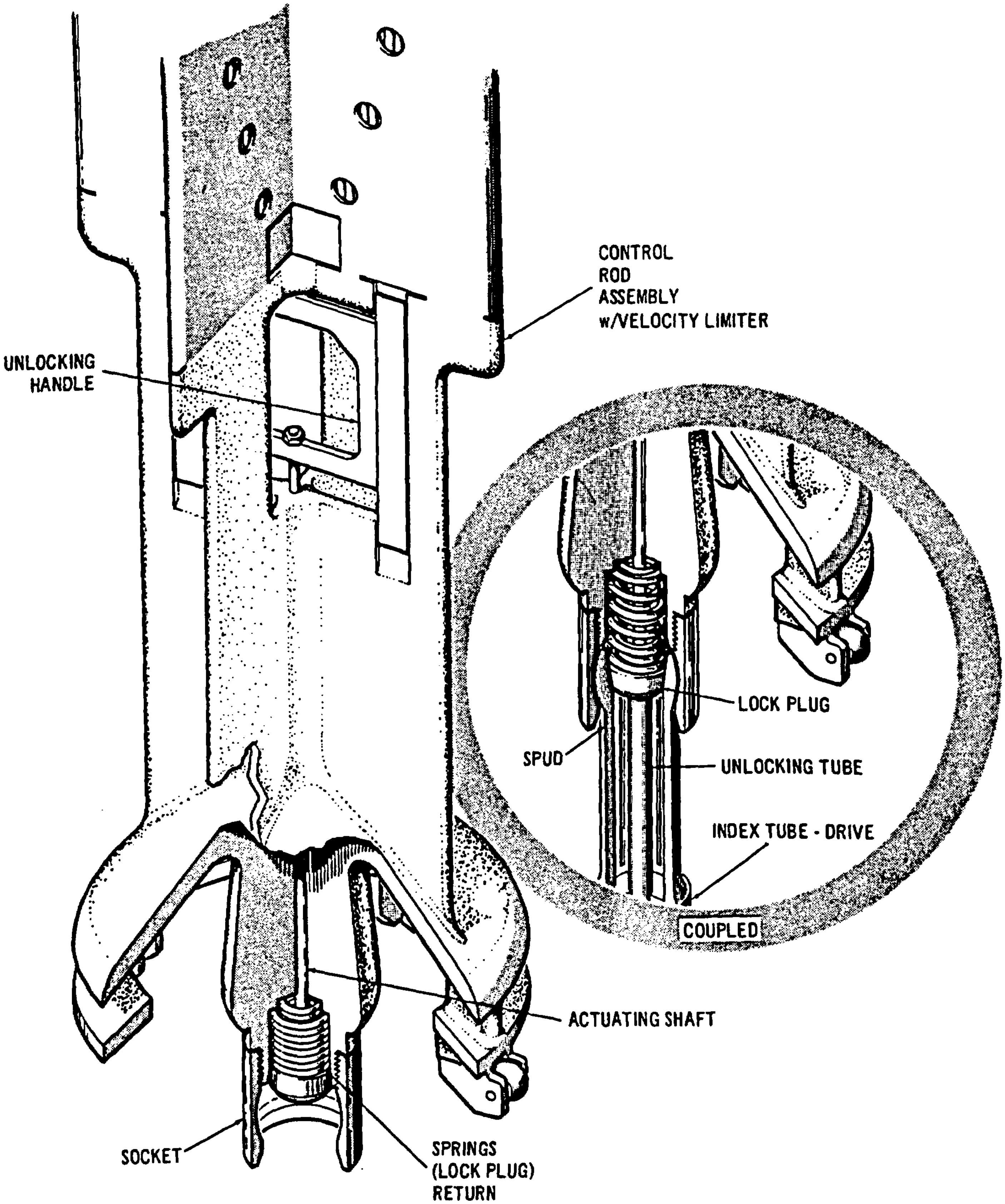
If the housing were to fail as described, the following sequence of events is foreseen:

a. The housing would separate from the reactor vessel.

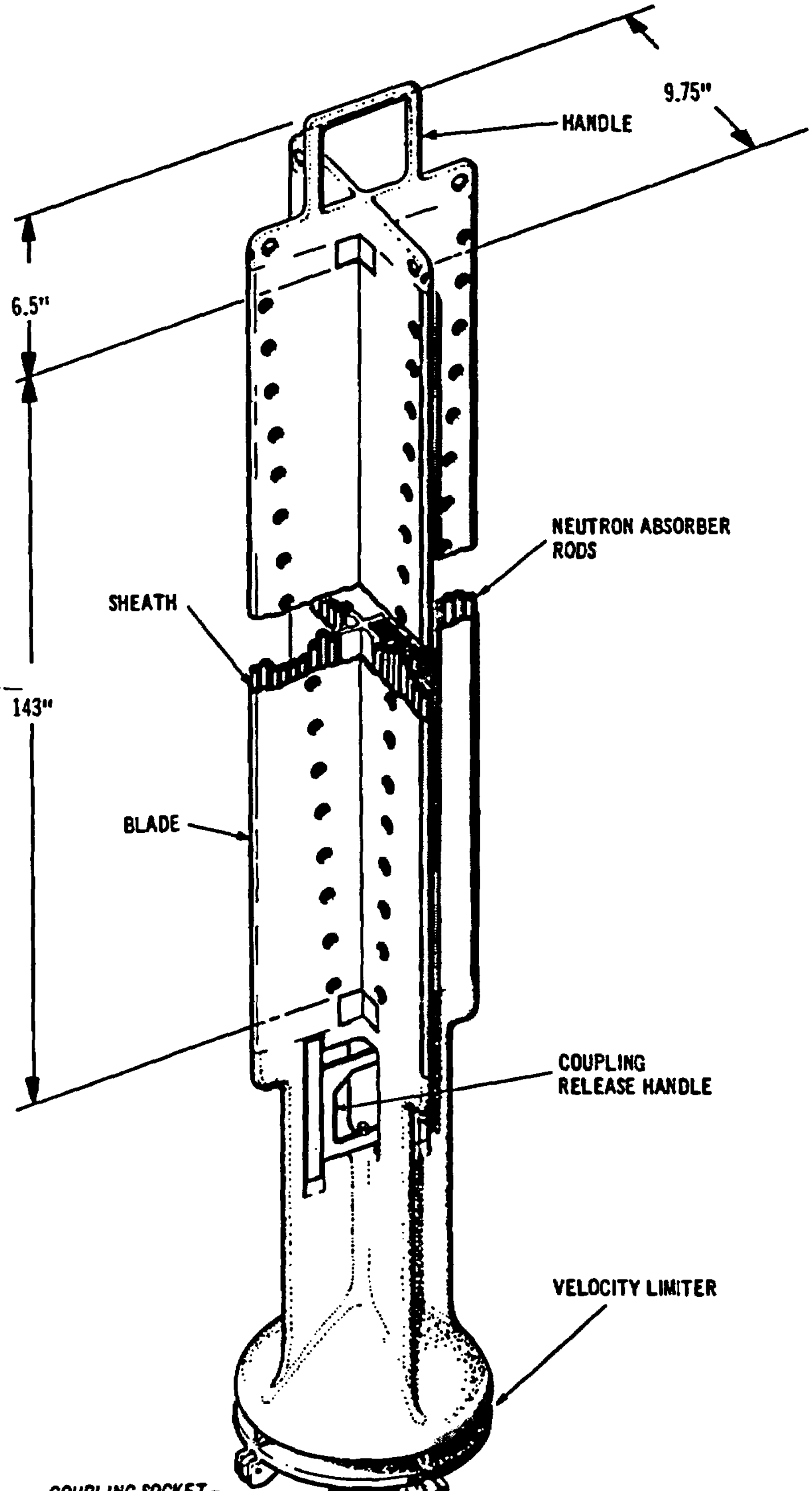
b. The CRD and housing would be blown downward against the support structure, by reactor pressure acting on the cross-sectional area of the housing and the drive.



GPU NUCLEAR CORPORATION OYSTER CREEK NUCLEAR GENERATING STATION UPDATED FINAL SAFETY ANALYSIS REPORT	CONTROL ROD DRIVE—CUTAWAY	
	REV. 0, 12/84	FIGURE 4.6-2



GPU NUCLEAR CORPORATION OYSTER CREEK NUCLEAR GENERATING STATION UPDATED FINAL SAFETY ANALYSIS REPORT	CONTROL ROD TO DRIVE COUPLING ISOMETRIC	
	REV. 0, 12/84	FIGURE 4.6-4



GPU NUCLEAR CORPORATION OYSTER CREEK NUCLEAR GENERATING STATION UPDATED FINAL SAFETY ANALYSIS REPORT	CONTROL ROD - ISOMETRIC	
	REV. 0, 12/84	FIGURE 4.6-7

Figure 3.5-1 Control Rod Assembly Isometric

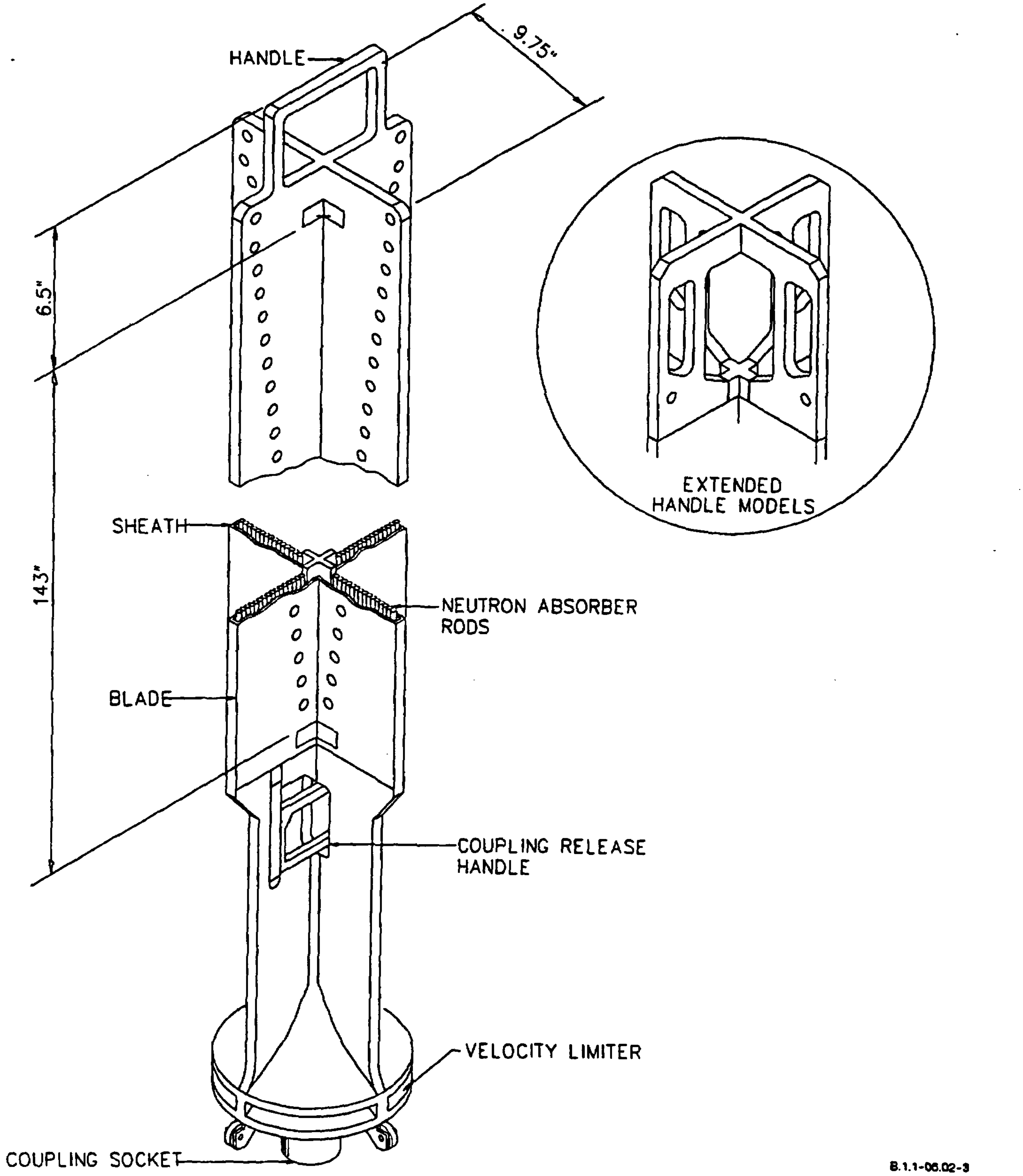


Figure 3.5-1a Duralife - 230 Control Blade

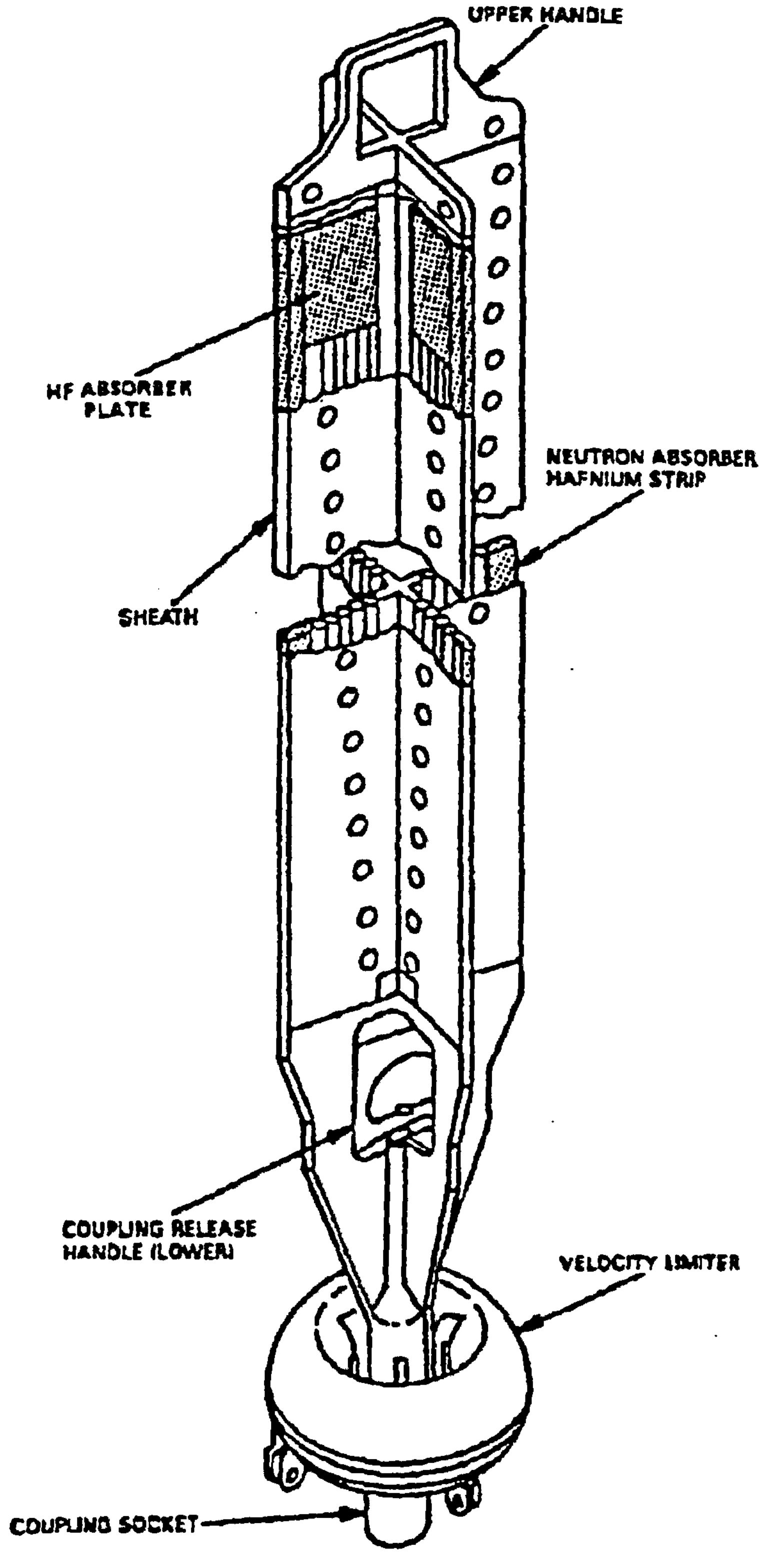
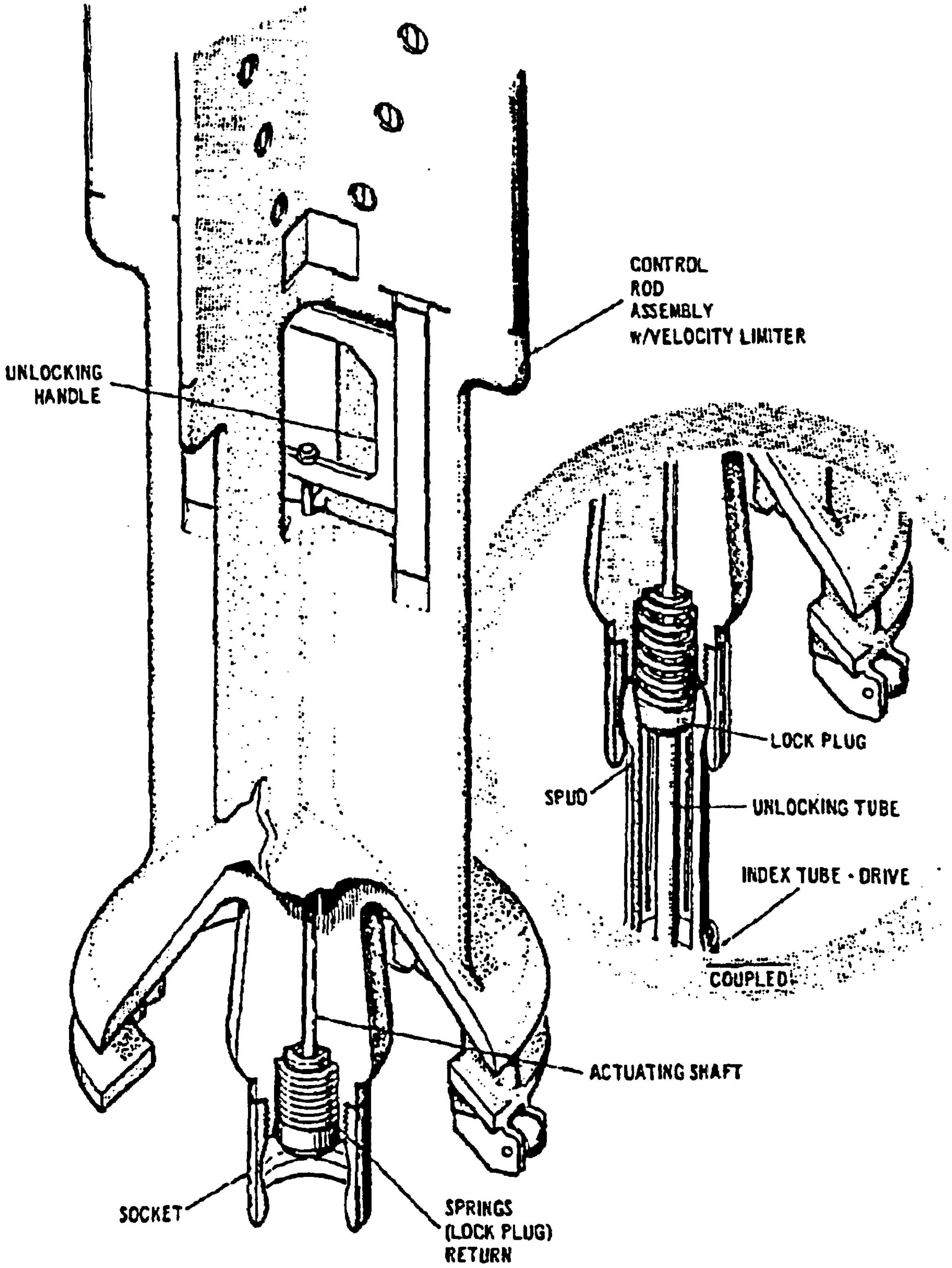
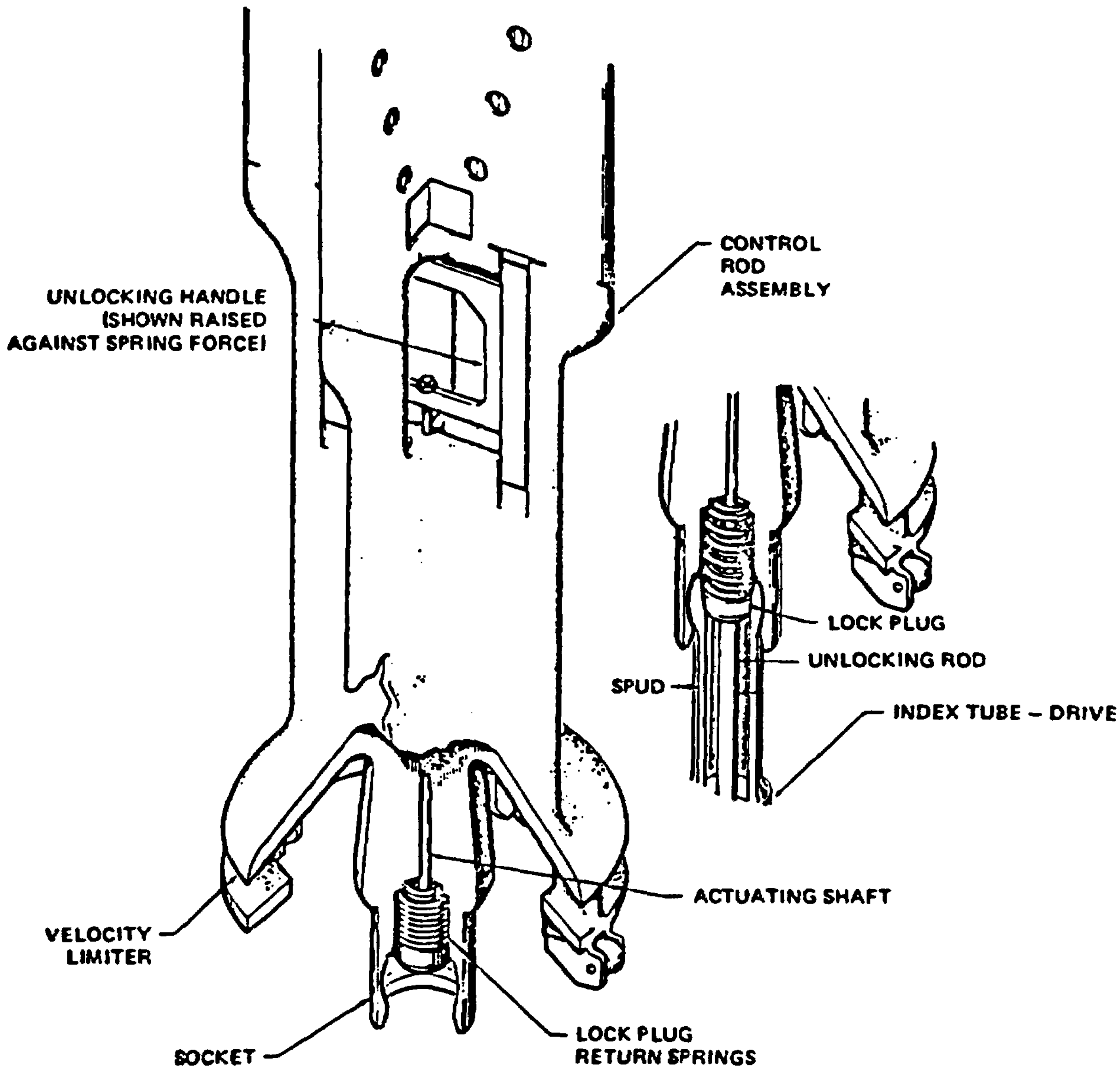


Figure 3.5-2 Control Rod Assembly and Drive Coupling Isometric

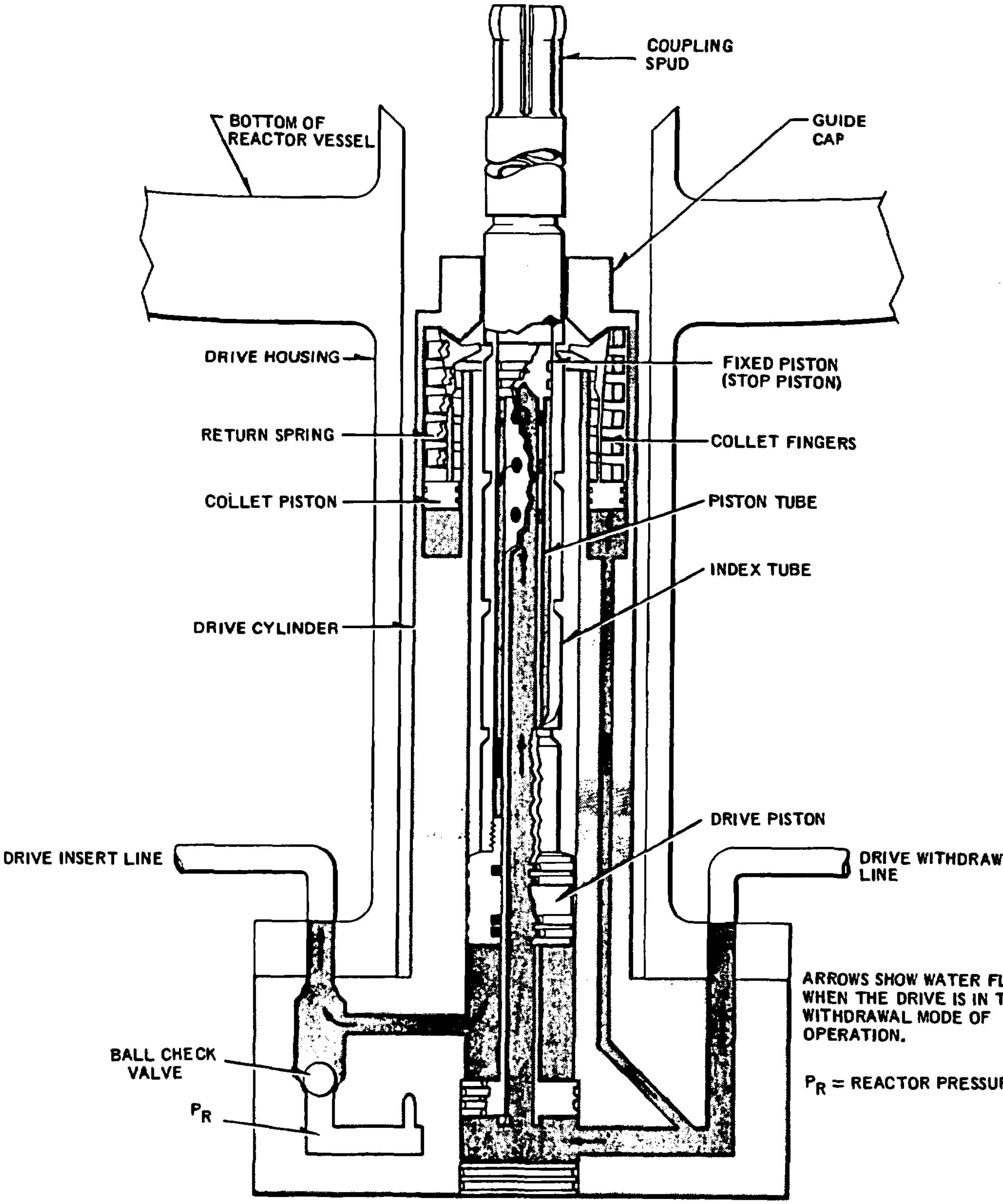




LIMERICK GENERATING STATION
UNITS 1 AND 2
UPDATED FINAL SAFETY ANALYSIS REPORT

CONTROL ROD TO CONTROL ROD DRIVE COUPLING

FIGURE 4.6-1



ARROWS SHOW WATER FLOW WHEN THE DRIVE IS IN THE WITHDRAWAL MODE OF OPERATION.

P_R = REACTOR PRESSURE

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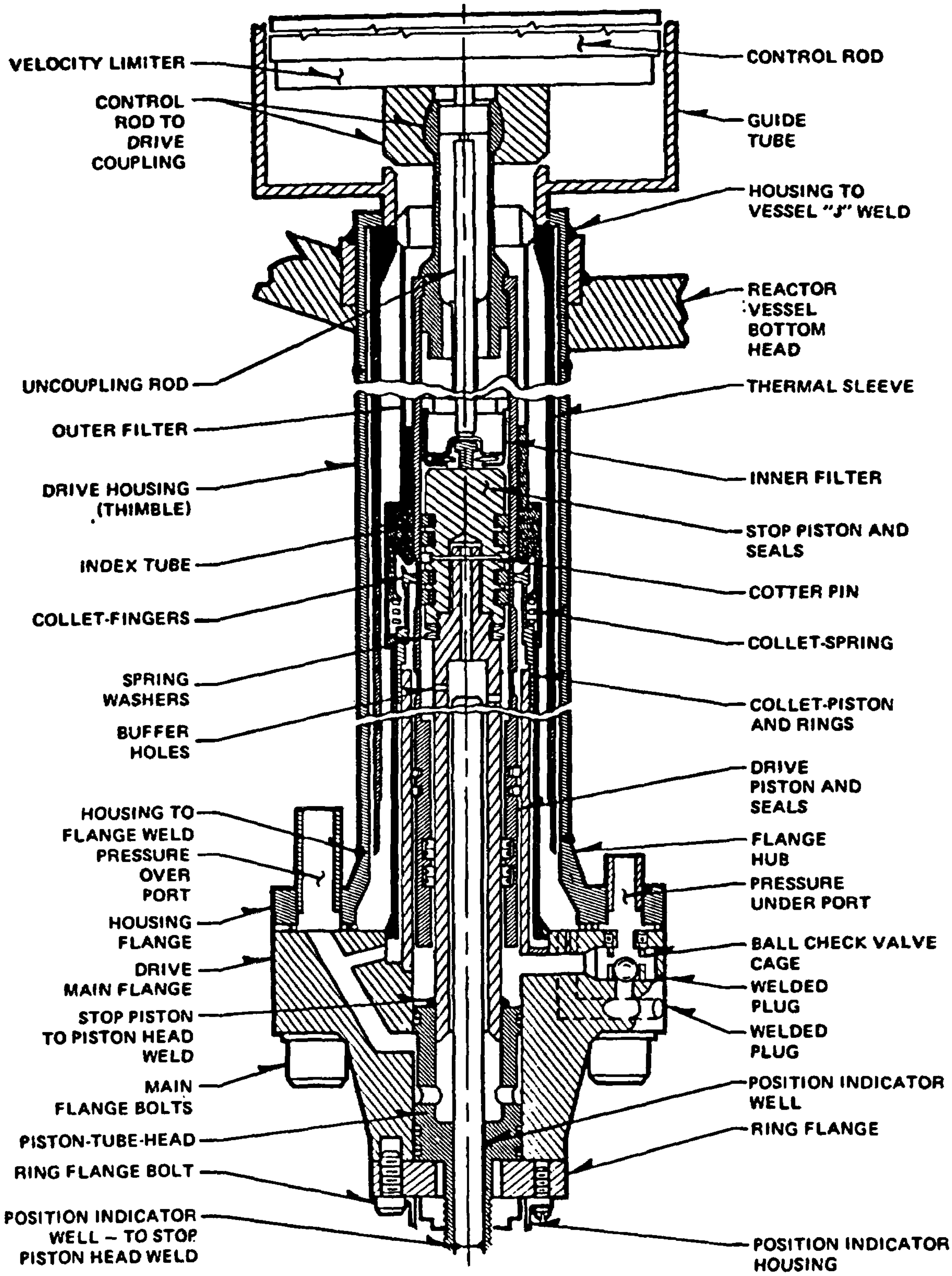
CONTROL ROD DRIVE UNIT

FIGURE 4.6-2



LIMERICK GENERATING STATION
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CONTROL ROD DRIVE SCHEMATIC
BWR/4 & 5

FIGURE 4.6-3

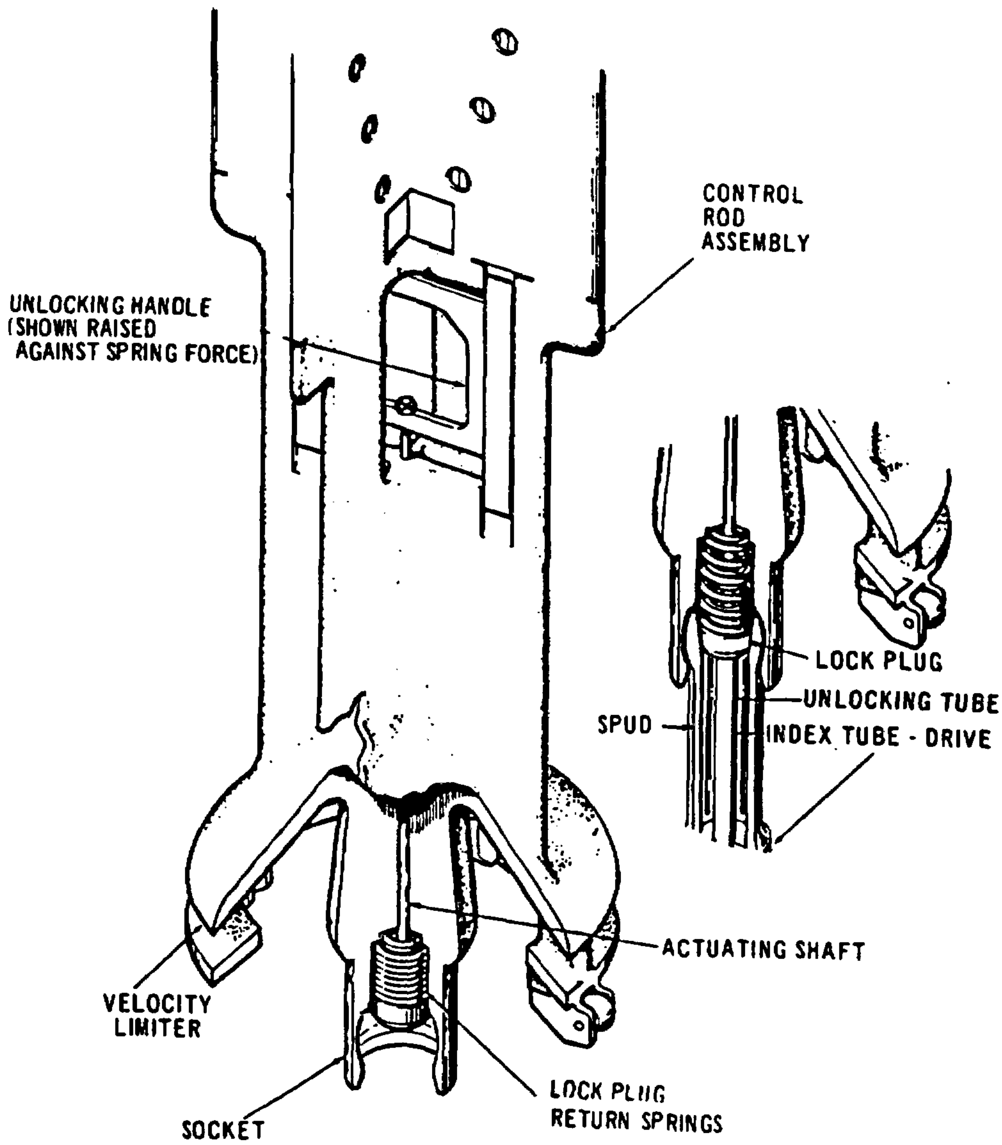

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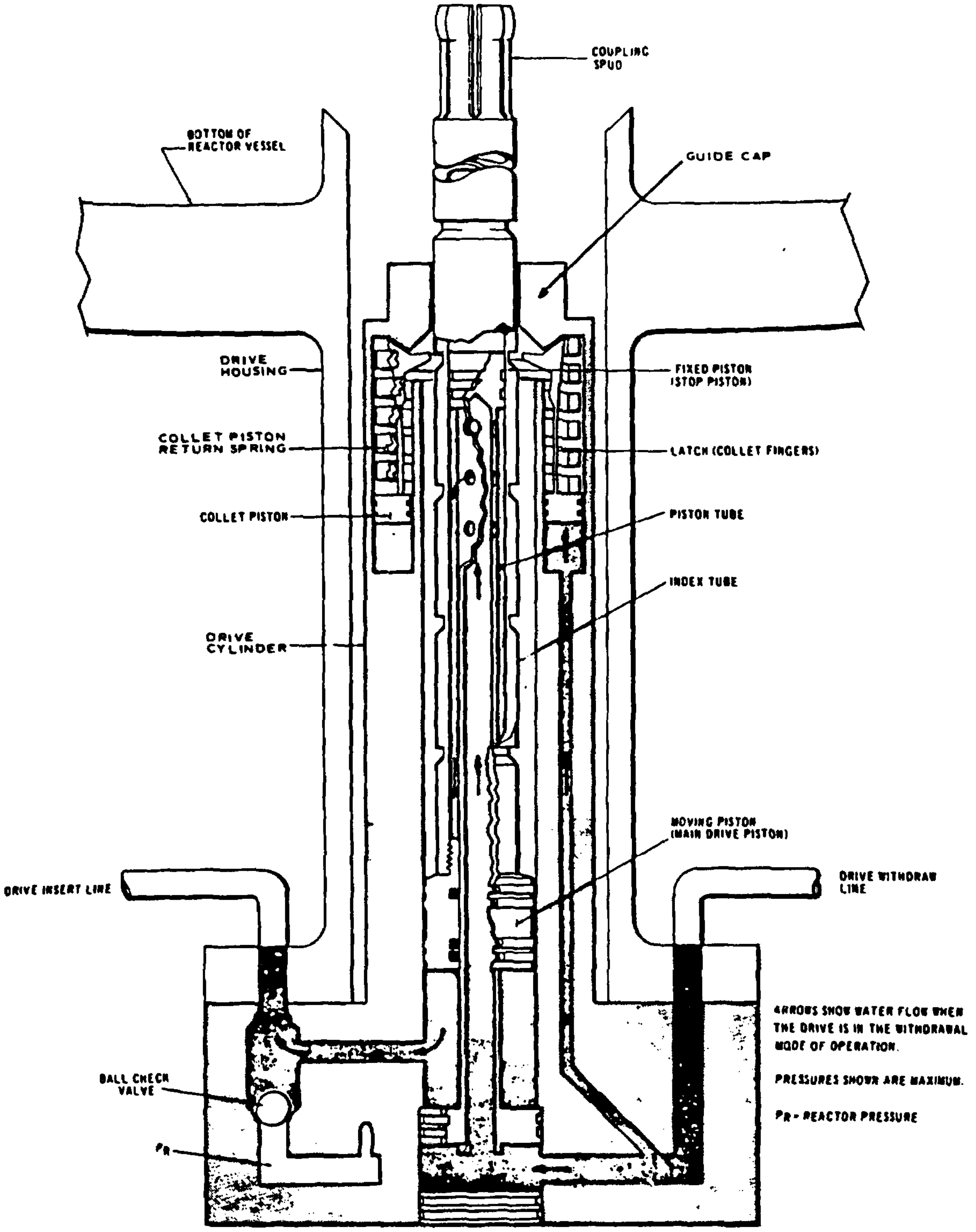
FIGURE 4.6-1

CONTROL ROD TO CONTROL ROD
DRIVE COUPLING



LA SALLE COUNTY STATION
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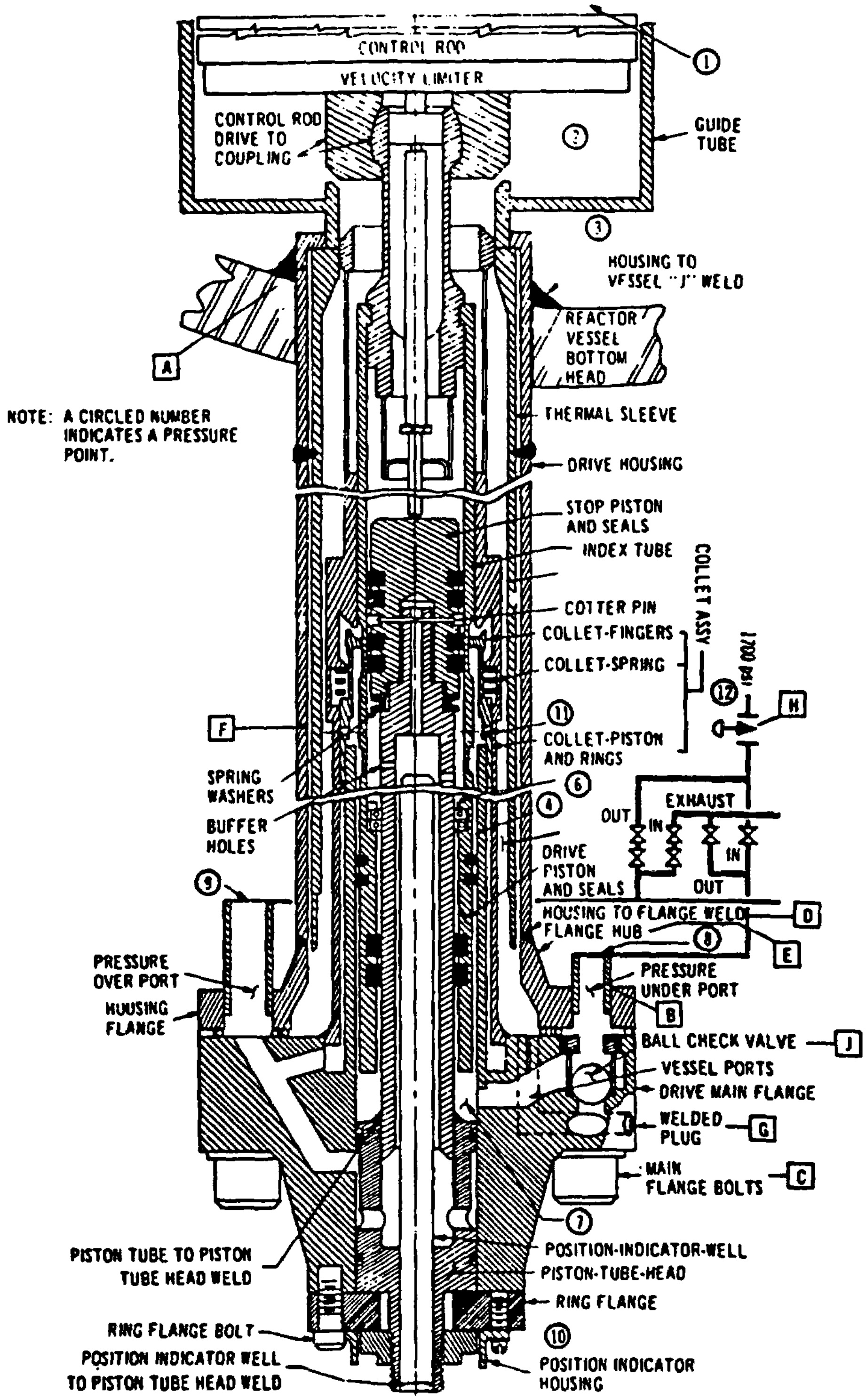
FIGURE 4.6-2

CONTROL ROD DRIVE UNIT



LA SALLE COUNTY STATION
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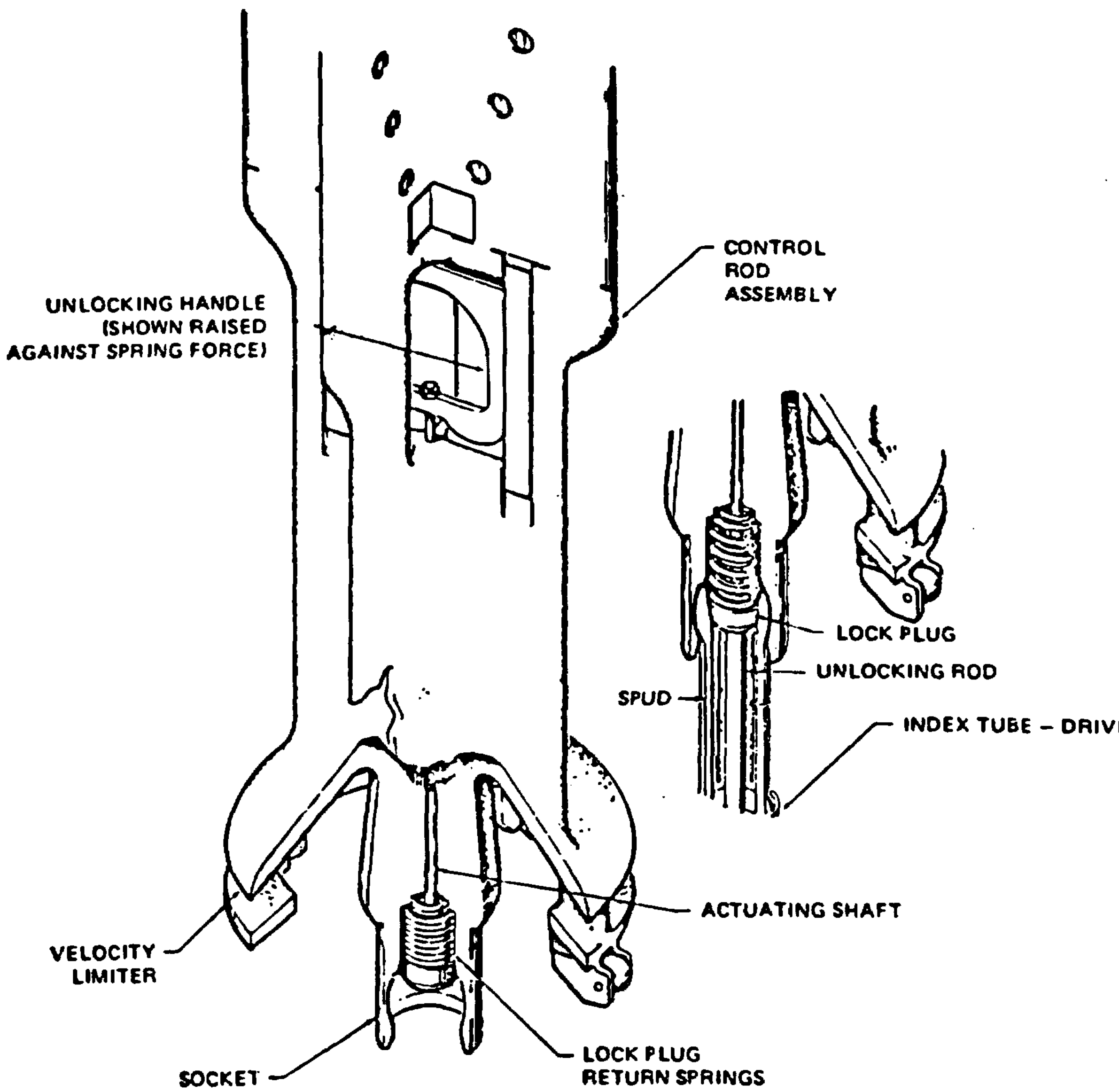
FIGURE 4.6-3

CONTROL ROD DRIVE UNIT (SCHEMATIC)

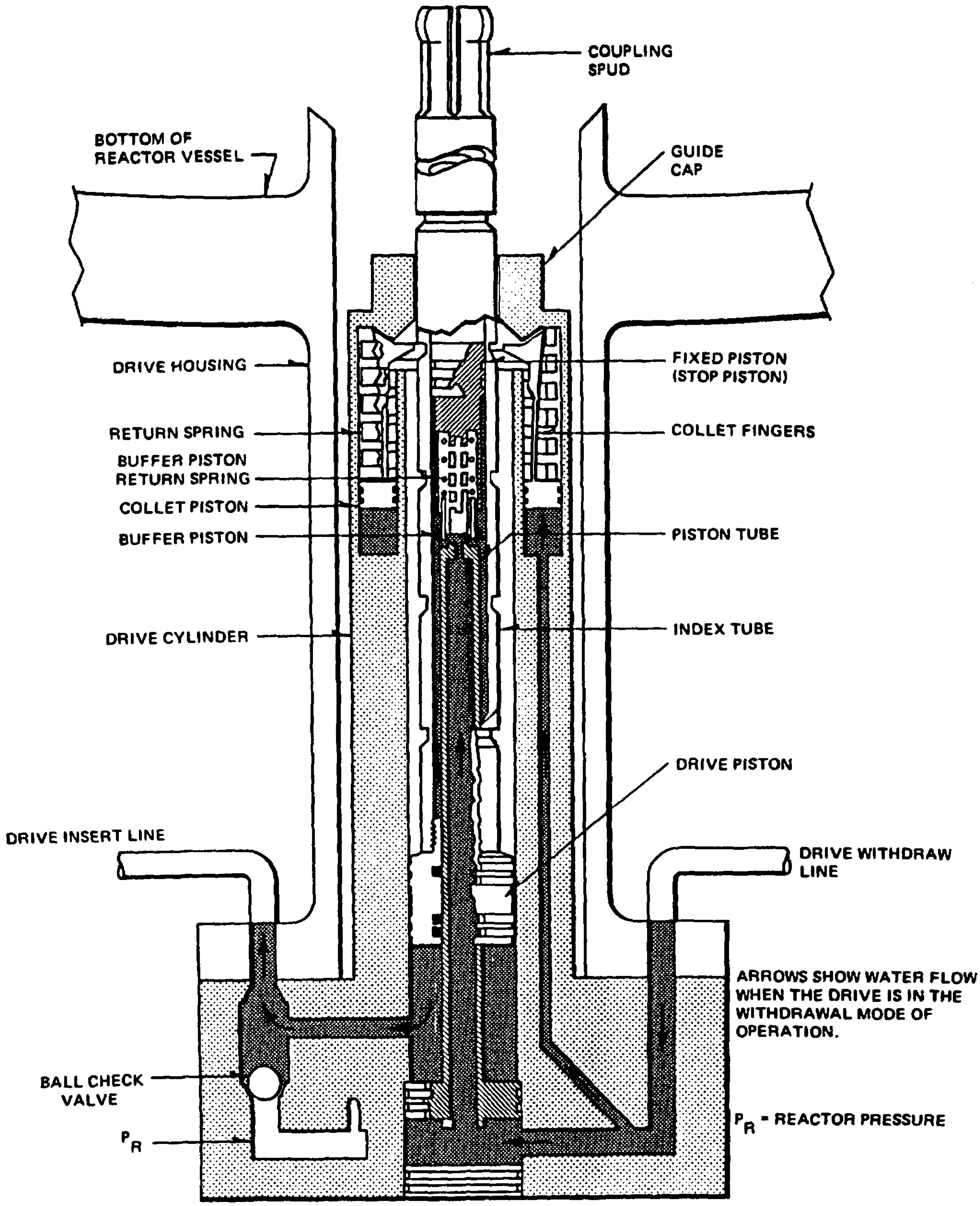
REV. 0 - APRIL 1984



PERRY NUCLEAR POWER PLANT
THE CLEVELAND ELECTRIC ILLUMINATING COMPANY

Control Rod to Control Rod Drive Coupling

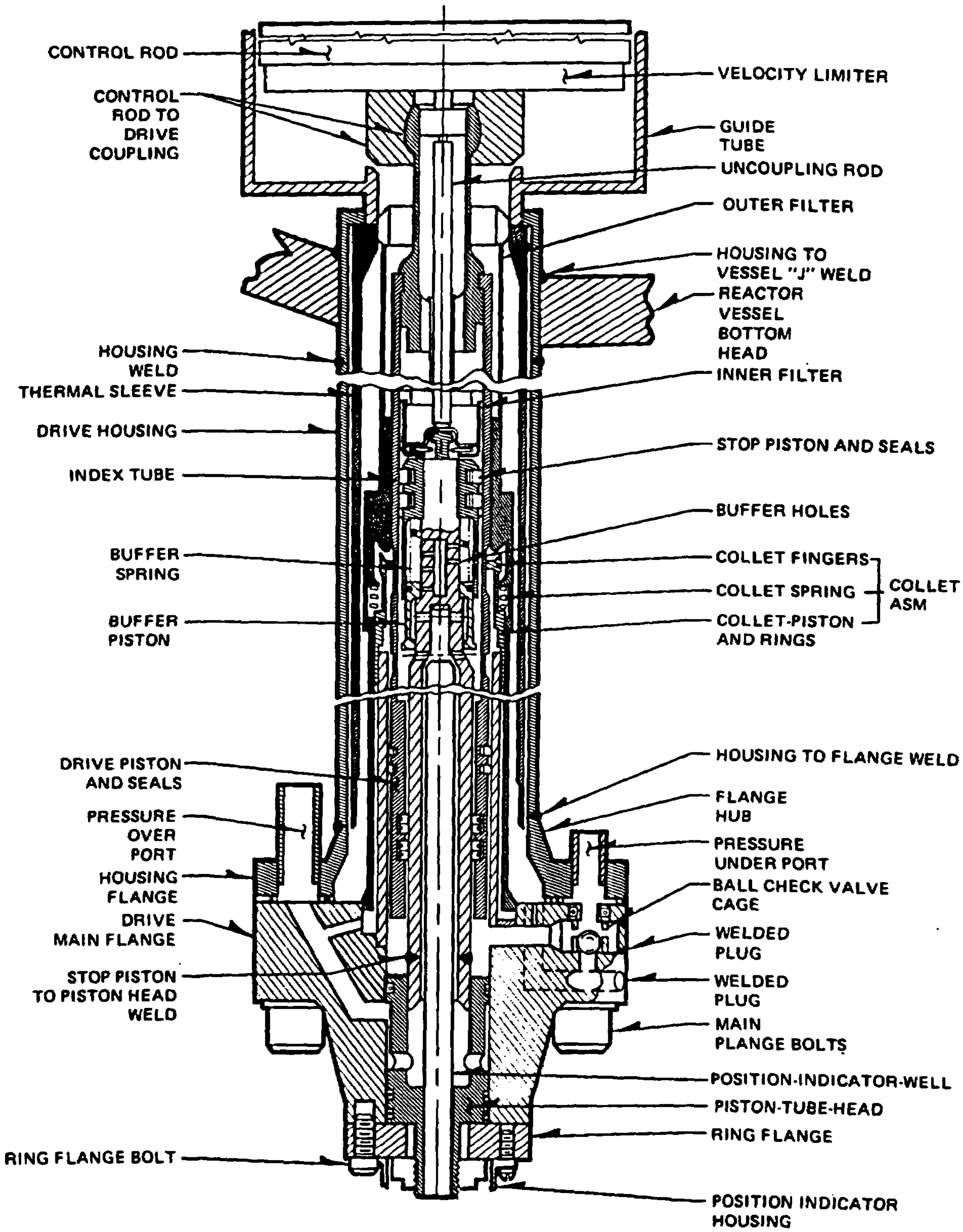
Figure 4.6-1



PERRY NUCLEAR POWER PLANT
THE CLEVELAND ELECTRIC ILLUMINATING COMPANY

Control Rod Drive Unit

Figure 4.6-2



PERRY NUCLEAR POWER PLANT
THE CLEVELAND ELECTRIC ILLUMINATING COMPANY

Control Rod Drive Schematic

Figure 4.6-3

Attachment 2

UFSAR Pages Related to the Response to RAI 2

4.6.2.3.2 Control Rod Drives

4.6.2.3.2.1 Evaluation of Scram Time

The rod scram function of the CRD system provides the negative reactivity insertion required by safety design basis as stated in Section 4.6.1.1.1. The scram time shown in the description is adequate as shown by the transient analyses of Chapter 15.

4.6.2.3.2.2 Analysis of Malfunction Relating to Rod Withdrawal

There are no known single malfunctions that cause the unplanned withdrawal of even a single control rod. However, if multiple malfunctions are postulated, studies show that an unplanned rod withdrawal can occur at withdrawal speeds that vary with the combination of malfunctions postulated. In all cases the subsequent withdrawal speeds are less than that assumed in the rod-drop accident analysis as discussed in Chapter 15. Therefore, the physical and radiological consequences of such rod withdrawals are less than those analyzed in the rod-drop accident.

4.6.2.3.2.2.1 Drive Housing Failure at Attachment Weld

The bottom head of the reactor vessel has a penetration for each CRD location. A drive housing is raised into position inside each penetration and is fastened by welding. The drive is raised into the drive housing and bolted to a flange at the bottom of the housing. The housing material is seamless, Type 304 stainless steel pipe with a minimum tensile strength of 75,000 psi. The basic failure considered here is a complete circumferential crack through the housing wall at an elevation just below the J-weld.

Static loads on the housing wall include the weight of the drive and the control rod, the weight of the housing below the J-weld, and the reactor pressure acting on the 6 inch diameter cross-sectional area of the housing and the drive. Dynamic loading results from the reaction force during drive operation.

If the housing were to fail as described, the following sequence of events is foreseen:

a. The housing would separate from the reactor vessel.

b. The CRD and housing would be blown downward against the support structure, by reactor pressure acting on the cross-sectional area of the housing and the drive.

c. The downward motion of the drive and associated parts would be determined by the gap between the bottom of the drive and the support structure, and by the deflection of the support structure under the load.

   1. In the current design, maximum deflection is approximately 3 inches.

   2. If the collet remains latched, no further control rod ejection would occur (Reference 4.6-1); the housing would not drop far enough to clear the reactor vessel penetration.

d. Reactor water would leak at a rate of approximately 180 gpm, through the 0.03 inch diameter clearance between the housing OD and reactor vessel penetration ID.

If the basic housing failure were to occur while the control rod is being withdrawn (this is a small fraction of the total drive operating time) and if the collet were to stay unlatched, the following sequence of events is foreseen:

a. The housing would separate from the reactor vessel.

b. The control rod, CRD, and housing would be blown downward against the CRD housing support.

c. Calculations indicate that the steady-state rod withdrawal velocity would be 0.3 ft/sec.

d. During withdrawal, pressure under the collet piston would be approximately 250 psi greater than the pressure over it; therefore, the collet would be held in the unlatched position until driving pressure was removed from the pressure-over port.

4.6.2.3.2.2.2 Rupture of Hydraulic Line(s) to Drive Housing Flange

There are three types of possible rupture of hydraulic lines to the drive housing flange: pressure-under (insert) line break; pressure-over (withdrawn) line break; and coincident breakage of both of these lines.

4.6.2.3.2.2.2.1 Pressure-Under (Insert) Line Break

For the case of a pressure-under (insert) line break, a partial or complete circumferential opening is postulated at or near the point where the line enters the housing flange. Failure is more likely to occur after another basic failure, wherein the drive housing or housing flange separates from the reactor vessel. Failure of the housing, however, does not necessarily lead directly to failure of the hydraulic lines.

If the pressure-under (insert) line were to fail and if the collet were latched, no control rod withdrawal would occur. There would be no pressure differential across the collet piston and, therefore, no tendency to unlatch the collet. Consequently, the associated control rod could not be withdrawn, but if reactor pressure is greater than 600 psig, it will insert on a scram signal.

The ball check valve is designed to seal off a broken pressure-under line by using reactor pressure to shift the check ball to its upper seat. If the ball check valve were prevented from seating, reactor water would leak to the reactor enclosure or containment. Because of the broken line, cooling water could not be supplied to the drive involved. Loss of cooling water would cause no immediate damage to the drive. However, prolonged exposure of the drive to temperatures at or near reactor temperature could lead to deterioration of material in the seals. High temperature would be indicated to the operator by the thermocouple in the position indicator probe, by high cooling water flow, and by operation of the containment sump pump.

If the basic line failure were to occur while the control rod is being withdrawn, the hydraulic force would not be sufficient to hold the collet open, and spring force normally would cause the collet to latch and stop rod withdrawal. However, if the collet were to remain open, calculations indicate that the steady-state control rod withdrawal velocity would be 2 ft/sec.

4.6.2.3.2.2.2.2 Pressure-Over (Withdrawn) Line Break

The case of the pressure-over (withdrawn) line breakage considers the complete breakage of the line at or near the point where it enters the housing flange. If the line were to break, pressure over the drive piston would drop from reactor pressure to atmospheric pressure. Any significant reactor pressure (approximately 600 psig or greater) would act on the bottom of the drive piston and fully insert the drive. Insertion would occur regardless of the operational mode at the time of the failure. After full insertion, reactor water would leak past the stop piston seals. This leakage would exhaust to the atmosphere through the broken pressure-over line. The leakage rate at 1000 psi reactor pressure is estimated to be 1-3 gpm; however, with the Graphitar seals of the stop piston removed, the leakage rate could be as high as 10 gpm, based on experimental measurements. If the reactor were hot, drive temperature would increase. This situation would be indicated to the reactor operator by the drift alarm, by the fully inserted drive, by a high drive temperature (annunciated in the control room), and by operation of the drywell sump pump.

4.6.2.3.2.2.2.3 Simultaneous Breakage of the Pressure-Over (Withdrawn) Pressure-Under (Insert) Lines

For the simultaneous breakage of the pressure-over (withdrawn) and pressure-under (insert) lines, pressures above and below the drive piston would drop to zero, and the ball check valve would close the broken pressure-under line. Reactor water would flow from the annulus outside the drive, through the vessel ports, and to the space below the drive piston. As in the case of pressure-over line breakage, the drive would then insert (at reactor pressure approximately 600 psi or greater) at a speed dependent on reactor pressure. Full insertion would occur regardless of the operational mode at the time of failure. Reactor water would leak past the drive seals and out the broken pressure-over line to the reactor enclosure or containment, as described above. Drive temperature would increase. Indication in the control room would include the drift alarm, the fully inserted drive, the high drive temperature annunciated in the control room, and the operation of the drywell sump pump.

4.6.2.3.2.2.3 All Drive Flange Bolts Fail in Tension

Each CRD is bolted to a flange at the bottom of a drive housing. The flange is welded to the drive housing. Bolts are made of AISI-4140 steel or AISI-4340 steel, with a minimum tensile strength of 125,000 psi. Each bolt has an allowable load capacity of at least 15,200 pounds. Capacity of the 8 bolts is at least 121,600 pounds. As a result of the reactor design pressure of 1250 psig, the major load on all 8 bolts is 30,400 pounds.

If a progressive or simultaneous failure of all bolts occurs, the drive separates from the housing. The control rod and the drive would be blown downward against the support structure. Impact velocity and support structure loading would be slightly less than that for drive housing failure, because reactor pressure would act on the drive's cross-sectional area only and the housing would remain attached to the reactor vessel. The drive would be isolated from the cooling water supply. Reactor water would flow downward past the velocity limiter piston, through the large drive filter, and into the annular space between the thermal sleeve and the drive. For worst case leakage calculations, the large filter is assumed to be deformed or swept out of the way so it would offer no significant flow restriction. At a point near the top of the annulus, where pressure would have dropped to 350 psi, the water would flash to steam and cause choke flow conditions. Steam would flow down the annulus and out the space between the housing and the drive flanges to the drywell. Steam formation would limit the leakage rate to approximately 840 gpm.

If the collet were latched, control rod ejection would be limited to the distance the drive can drop before coming to rest on the support structure. There would be no tendency for the collet to unlatch, because pressure below the collet piston would drop to zero. Pressure forces, in fact, exert 1435 pounds to hold the collet in the latched position.

If the bolts failed during control rod withdrawal, pressure below the collet piston would drop to zero. The collet, with 1650 pounds return force, would latch and stop rod withdrawal.

4.6.2.3.2.2.4 Weld Joining Flange-to-Housing Failure in Tension

The failure considered is a crack in or near the weld that joins the flange to the housing. This crack extends through the wall and completely around the housing. The flange material is forged, Type 304 stainless steel, with a minimum tensile strength of 75,000 psi. The housing material is seamless, Type 304 stainless steel pipe, with a minimum tensile strength of 75,000 psi. The conventional, full penetration weld of Type 308 stainless steel has a minimum tensile strength approximately the same as that for the parent metal. The design pressure and temperature are 1250 psig and 575°F. Reactor pressure acting on the cross-sectional area of the drive; the weight of the control rod, drive, and flange; and the dynamic reaction force during drive operation result in a maximum tensile stress at the weld of approximately 6000 psi.

If the basic flange-to-housing joint failure occurred, the flange and the attached drive would be blown downward against the support structure. The support structure loading would be slightly less than that for drive housing failure, because reactor pressure would act only on the drive cross-sectional area. Lack of differential pressure across the collet piston would cause the collet to remain latched and limit control rod motion to approximately 3 inches. Downward drive movement would be small; therefore, most of the drive would remain inside the housing. The pressure-under and pressure-over lines are flexible enough to withstand the small displacement and remain attached to the flange. Reactor water would follow the same leakage path described above for the flange bolt failure, except that exit to the drywell would be through the gap between the lower end of the housing and the top of the flange. Water would flash to steam in the annulus surrounding the drive. The leakage rate would be approximately 840 gpm.

If the basic failure were to occur during control rod withdrawal (a small fraction of the total operating time) and if the collet were held unlatched, the flange would separate from the housing. The drive and flange would be blown downward against the support structure. The calculated steady-state rod withdrawal velocity would be 0.13 ft/sec. Because pressure-under and pressure-over lines remain intact, driving water pressure would continue to the drive, and the normal exhaust line restriction would exist. The pressure below the velocity limiter piston would drop below normal as a result of leakage from the gap between the housing and the flange. This differential pressure across the velocity limiter piston would result in a net downward force of approximately 70 pounds. Leakage out of the housing would greatly reduce the pressure in the annulus surrounding the drive. Thus, the net downward force on the drive piston would be less than normal. The overall effect of these events would be to reduce rod withdrawal to approximately one-half of normal speed. With a 560-psi differential across the collet piston, the collet would remain unlatched; however, it should relatch as soon as the drive signal is removed.

4.6.2.3.2.2.5 Housing Wall Ruptures

This failure is a vertical split in the drive housing wall just below the bottom head of the reactor vessel. The flow area of the hole is considered equivalent to the annular area between the drive and the thermal sleeve. Thus, flow through this annular area, rather than flow through the hole in the housing, would govern leakage flow. The housing is made of Type 304 stainless steel seamless pipe, with a minimum tensile strength of 75,000 psi. The maximum hoop stress of 11,900 psi results primarily from the reactor design pressure (1250 psig) acting on the inside of the housing.

If such a rupture were to occur, reactor water would flash to steam, and leak through the hole in the housing to the drywell at approximately 1030 gpm. Choke flow conditions would exist, as described previously for the flange bolt failure. However, leakage flow would be greater because flow resistance would be less, that is, the leaking water and steam would not have to flow down the length of the housing to reach the drywell. A critical pressure of 350 psi causes the water to flash to steam.

There would be no pressure differential acting across the collet piston to unlatch the collet; but the drive would insert as a result of loss of pressure in the drive housing causing a pressure drop in the space above the drive piston.

If this failure occurred during control rod withdrawal, drive withdrawal would stop, but the collet would remain unlatched. The drive would be stopped by a reduction of the net downward force action on the drive line. The net force reduction would occur when the leakage flow of 1030 gpm reduces the pressure in the annulus outside the drive to approximately 540 psig, thereby reducing the pressure acting on top of the drive piston to the same value. A pressure differential of approximately 710 psi would exist across the collet piston and holds the collet unlatched as long as the operator holds the withdraw signal.

4.6.2.3.2.2.6 Flange Plug Blows Out

To connect the vessel ports with the bottom of the ball check valve, a hole of 0.75 inch diameter is drilled in the drive flange of this hole is sealed with a plug of 0.812 inch diameter and 0.25 inch thickness. A full penetration, Type 308 stainless steel weld holds the plug in place. The postulated failure is a full circumferential crack in this weld and subsequent blowout of the plug.

If the weld were to fail, the plug were to blow out, and the collet remained latched, there would be no control rod motion. There would be no pressure differential acting across the collet piston to unlatch the collet. Reactor water would leak past the velocity limiter piston, down the annulus between the drive and the thermal sleeve, through the vessel ports and drilled passage, and out the open plug hole to the drywell at approximately 320 gpm. Leakage calculations assume only liquid flows from the flange. Actually, hot reactor water would flash to steam, and choke flow conditions would exist. Thus, the expected leakage rate would be lower than the calculated value. Drive temperature would increase and initiate an alarm in the control room.

If this failure were to occur during control rod withdrawal and if the collet were to stay unlatched, calculations indicate that control rod withdrawal speed would be approximately 0.24 ft/sec. Leakage from the open plug hole in the flange would cause reactor water to flow downward past the velocity limiter piston. A small differential pressure across the piston would result in an insignificant driving force of approximately 10 pounds, tending to increase withdraw velocity.

A pressure differential of 295 psi across the collet piston would hold the collet unlatched as long as the driving signal was maintained.

Flow resistance of the exhaust path from the drive would be normal because the ball check valve would be seated at the lower end of its travel by pressure under the drive piston.

4.6.2.3.2.2.7 Ball Check Valve Plug Blows Out

As a means of access for machining the ball check valve cavity, a 1.25 inch diameter hole has been drilled in the flange forging. This hole is sealed with a plug with a 1.31 inch diameter and 0.38 inch thickness. A full penetration weld, utilizing Type 308 stainless steel filler, holds the plug in place. The failure postulated is a circumferential crack in this weld leading to a blowout of the plug.

If the plug were to blow out while the drive was latched, there would be no control rod motion. No pressure differential would exist across the collet piston to unlatch the collet. As in the previous failure, reactor water would flow past the velocity limiter, down the annulus between the drive and thermal sleeve, through the vessel ports and drilled passage, through the ball check valve cage and out the open plug hole to the drywell. The leakage calculations indicate that the flow rate would be 350 gpm. This calculation assumes liquid flow, but flashing of the hot reactor water to steam would reduce this rate to a lower value. Drive temperature would rapidly increase and initiate an alarm in the control room.

If the plug failure were to occur during control rod withdrawal, (it would not be possible to unlatch the drive after such a failure) the collet would relatch at the first locking groove. If the collet were to stick, calculations indicate the control rod withdrawal speed would be 11.8 ft/sec. There would be a large retarding force exerted by the velocity limiter due to a 35 psi pressure differential across the velocity limiter piston.

4.6.2.3.2.2.8 Drive/Cooling Water Pressure Control Valve Closure (Reactor Pressure, 0 psig)

The pressure to move a drive is generated by the pressure drop of practically the full system flow through the drive/cooling water pressure control valve. This valve is either a MOV or a standby manual valve; either one is adjusted to a fixed opening. The normal pressure drop across this valve develops a pressure 260 psi in excess of reactor pressure.

If the flow through the drive/cooling water pressure control valve were to be stopped, as by a valve closure or flow blockage, the drive pressure would increase to the shutoff pressure of the supply pump. The occurrence of this condition during withdrawal of a drive at zero vessel pressure will result in a drive pressure increase from 260 psig to no more than 1750 psig. Calculations indicate that the drive accelerates from 3 in/sec to approximately 6.5 in/sec. A pressure differential of 1670 psi across the collet piston would hold the collet unlatched. Flow would be upward, past the velocity limiter piston, but retarding force would be negligible. Rod movement would stop as soon as the driving signal was removed.

4.6.2.3.2.2.9 Ball Check Valve Fails to Close Passage to Vessel Ports

Should the ball check valve sealing the passage to the vessel ports be dislodged and prevented from reseating following the insert portion of a drive withdrawal sequence, water below the drive piston would return to the reactor through the vessel ports and the annulus between the drive and the housing rather than through the speed control valve. Because the flow resistance of this return path would be lower than normal, the calculated withdrawal speed would be 2 ft/sec. During withdrawal, differential pressure across the collet piston would be approximately 40 psi. Therefore, the collet would tend to latch and would have to stick open before continuous withdrawal at 2 ft/sec could occur. Water would flow upward past the velocity limiter piston, generating a small retarding force of approximately 120 pounds.

4.6.2.3.2.2.10 Hydraulic Control Unit Valve Failures

Various failures of the valves in the HCU can be postulated, but none could produce differential pressures approaching those described in the preceding paragraphs, and none alone could produce a high velocity withdrawal. Leakage through either one or both of the scram valves produces a pressure that tends to insert the control rod rather than to withdraw it. If the pressure in the SDV should exceed reactor pressure following a scram, a check valve in the line to the scram discharge header prevents this pressure from operating the drive mechanisms.

4.6.2.3.2.2.11 Collet Fingers Fail to Latch

The failure is presumed to occur when the drive withdraw signal is removed. If the collet fails to latch, the drive continues to withdraw at a fraction of the normal speed. This assumption is made because there is no known means for the collet fingers to become unlocked without some initiating signal. Because the collet fingers will not cam open under a load, accidental application of a down signal does not unlock them. (The drive must be given a short insert signal to unload the fingers and cam them open before the collet can be driven to the unlock position.) If the drive withdrawal valve fails to close following a rod withdrawal, the collet would remain open and the drive would continue to move at a reduced speed.

4.6.2.3.2.2.12 Withdrawal Speed Control Valve Failure

Normal withdrawal speed is determined by differential pressures in the drive and is set for a nominal value of 3 in/sec. Withdrawal speed is maintained by the pressure regulating system and is independent of reactor vessel pressure. Tests show that accidental opening of the speed control valve to the fully open position produces a velocity of approximately 6 in/sec.

The CRD system prevents unplanned rod withdrawal, and it has been shown above that only multiple failures in a drive unit and in its control unit could cause an unplanned rod withdrawal.

4.6.2.3.2.3 Scram Reliability

High scram reliability is the result of a number of features of the CRD system. For example:

a. Two reliable sources of scram energy are used to insert each control rod: individual accumulators at low reactor pressure, and the reactor vessel pressure itself at power.

b. Each drive mechanism has its own scram valves and two scram pilot valves, so only one drive can be affected if a scram valve fails to open. Both pilot valves must be de-energized to initiate a scram.

c. The RPS and the HCUs are designed so that the scram signal and mode of operation override all others.

d. The collet assembly and index tube are designed so they do not restrain or prevent control rod insertion during scram.

e. The SDV is monitored for accumulated water and the reactor scrams before the volume is reduced to a point that could interfere with a scram.

4.6.2.3.2.4 Control Rod Support and Operation

As described above, each control rod is independently supported and controlled as required by safety design bases.

4.6.2.3.3 Control Rod Drive Housing Supports

Downward travel of the CRD housing and its control rod following the postulated housing failure equals the sum of these distances: the compression of the disc springs under dynamic loading, and the initial gap between the grid and the bottom contact surface of the CRD flange. If the reactor were cold and pressurized, the downward motion of the control rod would be limited to the spring compression (approximately 2 inches) plus a gap of approximately 3/4" ± 1/4". If the reactor were hot and pressurized, the gap would be approximately 1/2" ± 1/4" and the spring compression would be slightly less than in the cold condition. In either case, the control rod movement following a housing failure is substantially limited below one drive "notch" movement (6 inches). Sudden withdrawal of any control rod, through a distance of one drive notch at any position in the core, does not produce a transient sufficient to damage any radioactive material barrier.

The CRD housing supports are in place during power operation and when the nuclear system is pressurized. If a control rod is ejected during shutdown, the reactor remains subcritical because it is designed to remain subcritical with any one control rod fully withdrawn at any time.

At plant operating temperature, a gap of approximately 1/2" ± 1/4" exists between the CRD housing and the supports. At lower temperatures the gap is greater. Because the supports do not contact any of the CRD housing, except during the postulated accident condition, vertical contact stresses are prevented. Inspection and testing are discussed in Section 4.6.3.2.

4.6.2.3.1.9 Evaluation of Control Rod Velocity Limiter

The control rod velocity limiter limits the free fall velocity of the control rod to a value that cannot result in nuclear system process barrier damage. This velocity is evaluated by the rod drop accident analysis in Chapter 15.

4.6.2.3.2 Control Rod Drives

4.6.2.3.2.1 Evaluation of Scram Time

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The rod scram function of the CRD system provides the negative reactivity insertion required by safety design basis No. 1 in Section 4.6.1.1.1.1. The scram time shown in the description (Section 4.6.1.1.2.5.3) is adequate as shown by the transient analyses of Chapter 15.

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4.6.2.3.2.2 Analysis of Malfunction Relating to Rod Withdrawal

There are no known single malfunctions that cause the unplanned withdrawal of even a single control rod. However, if multiple malfunctions are postulated, studies show that an unplanned rod withdrawal can occur at withdrawal speeds that vary with the combination of malfunctions postulated. In all cases the subsequent withdrawal speeds are less than those assumed in the rod drop accident analysis as discussed in Chapter 15. Therefore, the physical and radiological consequences of such rod withdrawals are less than those analyzed in the rod drop accident.

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The calculated values shown in the following postulated malfunction events may increase slightly (up to approximately 5% to 10%) when operating at a power uprate reactor pressure condition.

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4.6.2.3.2.2.1 Drive Housing Fails at Attachment Weld

The bottom head of the reactor vessel has a penetration for each CRD location. A drive housing is raised into position inside each penetration and fastened by welding. The drive is raised into the drive housing and bolted to a flange at the bottom of the housing.

The CRD housing material at the vessel penetration is seamless, Type Inconel 600 tubing with a minimum tensile strength of 80,000 psi, and Type 304 stainless steel pipe below the vessel with a minimum strength of 75,000 psi. The basic failure considered here is a complete circumferential crack through the housing wall at an elevation just below the J-weld.

Static loads on the housing wall include the weight of the drive and the control rod, the weight of the housing below the J-weld, and the reactor pressure acting on the 6-in diameter cross-sectional area of the housing and the drive.

Dynamic loading results from the reaction force during drive operation.

If the housing were to fail as described, the following sequence of events is foreseen. The housing would separate from the vessel. The CRD and housing would be blown downward against the support structure by reactor pressure acting on the cross-sectional area of the housing and the drive. The downward motion of the drive and associated parts would be determined by the gap between the bottom of the drive and the support structure and by the deflection of the support structure under load. In the current design, maximum deflection is approximately 3 in. If the collet were to remain latched, no further control rod ejection would occur[1], the housing would not drop far enough to clear the vessel penetration, and reactor water would leak at a rate of approximately 180 gpm through the 0.03-in diametral clearance between the housing and the vessel penetration.

If the basic housing failure were to occur while the control rod is being withdrawn (this is a small fraction of the total drive operating time) and if the collet were to stay unlatched, the following sequence of events is foreseen. The housing would separate from the vessel and the drive and housing would be blown downward against the CRD housing support. Calculations indicate that the steady-state rod withdrawal velocity would be 0.3 ft/sec. During withdrawal, pressure under the collet piston would be approximately 250 psi greater than the pressure over it. Therefore, the collet would be held in the unlatched position until driving pressure was removed from the pressure-over port.

4.6.2.3.2.2.2 Rupture of Hydraulic Line(s) to Drive Housing Flange

There are three types of possible rupture of hydraulic lines to the drive housing flange: 1) pressure-under (insert) line break, 2) pressure-over (withdrawn) line break, and 3) coincident breakage of both these lines.

4.6.2.3.2.2.2.1 Pressure-Under (Insert) Line Break

For the case of a pressure-under (insert) line break, a partial or complete circumferential opening is postulated at or near the point where the line enters the housing flange. Failure is more likely to occur after another basic failure wherein the drive housing or housing flange separates from the reactor vessel. Failure of the housing, however, does not necessarily lead directly to failure of the hydraulic lines.

If the pressure-under (insert) line were to fail and if the collet were latched, no control rod withdrawal would occur. There would be no pressure differential across the collet piston and, therefore, no tendency to unlatch the collet. Consequently, the associated control rod could not be withdrawn, but if reactor pressure is greater than 600 psig, it will insert on a scram signal.

The ball check valve is designed to seal off a broken pressure-under line by using reactor pressure to shift the check ball to its upper seat. If the ball check valve were prevented from seating, reactor water would leak to the containment. Because of the broken line, cooling water could not be supplied to the drive involved. Loss of cooling water would cause no immediate damage to the drive. However, prolonged exposure of the drive to temperatures at or near reactor temperature could lead to deterioration of material in the seals. High temperature would be indicated to the operator by the thermocouple in the position indicator probe. A second indication would be high cooling water flow.

If the basic line failure were to occur while the control rod is being withdrawn the hydraulic force would not be sufficient to hold the collet open, and spring force normally would cause the collet to latch and stop rod withdrawal. However, if the collet were to remain open, calculations indicate that the steady-state control rod withdrawal velocity would be 2 ft/sec.

4.6.2.3.2.2.2.2 Pressure-Over (Withdrawn) Line Break

The case of the pressure-over (withdrawn) line breakage considers the complete breakage of the line at or near the point where it enters the housing flange. If the line were to break, pressure over the drive piston would drop from reactor pressure to atmospheric pressure. Any significant reactor pressure (approximately 600 psig or greater) would act on the bottom of the drive piston and fully insert the drive. Insertion would occur regardless of the operational mode at the time of the failure. After full insertion, reactor water would leak past the stop piston seals. This leakage would exhaust to the containment through the broken pressure-over line. The leakage rate at 1,000 psi reactor pressure is estimated to be 1 to 3 gpm, however with the graphitar seals of the stop piston removed, the leakage rate could be as high as 10 gpm, based on experimental measurements. If the reactor were hot, drive temperature would increase. This situation would be indicated to the reactor operator by the drift alarm, by the fully inserted drive, by a high drive temperature annunciated in the main control room, by sump water level change detected by the high-sensitivity drywell-sump leak detection system, and by operation of the drywell sump pump.

4.6.2.3.2.2.2.3 Simultaneous Breakage of the Pressure-Over (Withdrawn) and Pressure-Under (Insert) Lines

For the simultaneous breakage of the pressure-over (withdrawn) and pressure-under (insert) lines, pressures above and below the drive piston would drop to zero, and the ball check valve would close the broken pressure-under line. Reactor water would flow from the annulus outside the drive, through the vessel ports, and to the space below the drive piston. As in the case of pressure-over line breakage, the drive would then insert (at reactor pressure approximately 600 psig or greater) at a speed dependent on reactor pressure. Full insertion would occur regardless of the operational mode at the time of failure. Reactor water would leak past the drive seals and out the broken pressure-over line to the containment, as described above. Drive temperature would increase. Indication in the main control room would include the drift alarm, the fully inserted drive, the high drive temperature annunciated in the main control room, and operation of the drywell sump pump.

4.6.2.3.2.2.3 All Drive Flange Bolts Fail in Tension

Each CRD is bolted to a flange at the bottom of a drive housing. The flange is welded to the drive housing. Bolts are made of AISI-4140 steel with a minimum tensile strength of 125,000 psi. Each bolt has an allowable load capacity of 15,200 lb. Capacity of the eight bolts is 121,600 lb. As a result of the reactor design pressure of 1,250 psig, the major load on all eight bolts is 30,400 lb.

If a progressive or simultaneous failure of all bolts were to occur, the drive would separate from the housing. The control rod and the drive would be blown downward against the support structure. Impact velocity and support structure loading would be slightly less than that for drive housing failure, because reactor pressure would act on the drive cross-sectional area only and the housing would remain attached to the reactor vessel. The drive would be isolated from the cooling water supply. Reactor water would flow downward past the velocity limiter piston, through the large drive filter, and into the annular space between the thermal sleeve and the drive. For worst-case leakage calculations, the large filter is assumed to be deformed or swept out of the way so it would offer no significant flow restriction. At a point near the top of the annulus, where pressure would have dropped to 350 psi, the water would flash to steam and cause choke-flow conditions. Steam would flow down the annulus and out the space between the housing and the drive flanges to the drywell. Steam formation would limit the leakage rate to approximately 840 gpm.

If the collet were latched, control rod ejection would be limited to the distance the drive can drop before coming to rest on the support structure. There would be no tendency for the collet to unlatch, because pressure below the collet piston would drop to zero. Pressure forces, in fact, exert 1,435 lb to hold the collet in the latched position.

If the bolts failed during control rod withdrawal, pressure below the collet piston would drop to zero. The collet, with 1,650 lb return force, would latch and stop rod withdrawal.

4.6.2.3.2.2.4 Weld Joining Flange to Housing Fails in Tension

The failure considered is a crack in or near the weld that joins the flange to the housing. This crack extends through the wall and completely around the housing. The flange material is forged, Type 304 stainless steel, with a minimum tensile strength of 75,000 psi. The housing material is seamless, Type 304 stainless steel pipe, with a minimum tensile strength of 75,000 psi. The conventional, full penetration weld of Type 308 stainless steel has a minimum tensile strength approximately the same as that for the parent metal. The design pressure and temperature are 1,250 psig and 575°F. Reactor pressure acting on the cross-sectional area of the drive; the weight of the control rod, drive, and flange; and the dynamic reaction force during drive operation result in a maximum tensile stress at the weld of approximately 5,100 psi.

If the basic flange-to-housing joint failure occurred, the flange and the attached drive would be blown downward against the support structure. The support structure loading would be slightly less than that for drive housing failure, because reactor pressure would act only on the drive cross-sectional area. Lack of differential pressure across the collet piston would cause the collet to remain latched and limit control rod motion to approximately 3 in. Downward drive movement would be small; therefore, most of the drive would remain inside the housing. The pressure-under and pressure-over lines are flexible enough to with- stand the small displacement and remain attached to the flange. Reactor water would follow the same leakage path described above for the flange-bolt failure, except that exit to the drywell would be through the gap between the lower end of the housing and the top of the flange. Water would flash to steam in the annulus surrounding the drive. The leakage rate would be approximately 840 gpm.

If the basic failure were to occur during control rod withdrawal (a small fraction of the total operating time) and if the collet were held unlatched, the flange would separate from the housing. The drive and flange would be blown downward against the support structure. The calculated steady-state rod withdrawal velocity would be 0.13 ft/sec. Because pressure-under and pressure-over lines remain intact, driving water pressure would continue to the drive, and the normal exhaust line restriction would exist. The pressure below the velocity limiter piston would drop below normal as a result of leakage from the gap between the housing and the flange. This differential pressure across the velocity limiter piston would result in a net downward force of approximately 70 lb. Leakage out of the housing would greatly reduce the pressure in the annulus surrounding the drive. Thus, the net downward force on the drive piston would be less than normal. The overall effect of these events would be to reduce rod withdrawal to approximately one-half of normal speed. With a 560-psi differential across the collet piston, the collet would remain unlatched; however, it should relatch as soon as the drive signal is removed.

4.6.2.3.2.2.5 Housing Wall Ruptures

This failure is a vertical split in the drive housing wall just below the bottom head of the reactor vessel. The flow area of the hole is considered equivalent to the annular area between the drive and the thermal sleeve. Thus, flow through this annular area, rather than flow through the hole in the housing, would govern leakage flow. The CRD housing is made of Inconel 600 seamless tubing (at the penetration to the vessel), with a minimum tensile strength of 80,000 psi, and of Type 304 stainless steel seamless pipe below the vessel with a minimum tensile strength of 75,000 psi. The maximum hoop stress of 9,000 psi results primarily from the reactor design pressure (1,250 psig) acting on the inside of the housing.

If such a rupture were to occur, reactor water would flash to steam and leak through the hole in the housing to the drywell at approximately 1,030 gpm. Choke-flow conditions would exist, as described previously for the flange-bolt failure. However, leakage flow would be greater because flow resistance would be less, that is, the leaking water and steam would not have to flow down the length of the housing to reach the drywell. A critical pressure of 350 psi causes the water to flash to steam.

There would be no pressure differential acting across the collet piston to unlatch the collet; but the drive would insert as a result of loss of pressure in the drive housing causing a pressure drop in the space above the drive piston.

If this failure occurred during control rod withdrawal, drive withdrawal would stop, but the collet would remain unlatched. The drive would be stopped by a reduction of the net downward force action on the drive line. The net force reduction would occur when the leakage flow of 1,030 gpm reduces the pressure in the annulus outside the drive to approximately 540 psig, thereby reducing the pressure acting on top of the drive piston to the same value. A pressure differential of approximately 710 psi would exist across the collet piston and hold the collet unlatched as long as the operator held the withdraw signal.

4.6.2.3.2.2.6 Flange Plug Blows Out

To connect the vessel ports with the bottom of the ball check valve, a hole of 3/4-in diameter is drilled in the drive flange. The outer end of this hole is sealed with a plug of 0.812-in diameter and 0.25-in thickness. A full-penetration, Type 308 stainless steel weld holds the plug in place. The postulated failure is a full circumferential crack in this weld and subsequent blowout of the plug.

If the weld were to fail, the plug were to blow out, and the collet remained latched, there would be no control rod motion. There would be no pressure differential acting across the collet piston to unlatch the collet. Reactor water would leak past the velocity limiter piston, down the annulus between the drive and the thermal sleeve, through the vessel ports and drilled passage, and out the open plug hole to the drywell at approximately 320 gpm. Leakage calculations assume only liquid flows from the flange. Actually, hot reactor water would flash to steam, and choke-flow conditions would exist. Thus, the expected leakage rate would be lower than the calculated value. Drive temperature would increase and initiate an alarm in the main control room.

If this failure were to occur during control rod withdrawal and if the collet were to stay unlatched, calculations indicate that control rod withdrawal speed would be approximately 0.24 ft/sec. Leakage from the open plug hole in the flange would cause reactor water to flow downward past the velocity limiter piston. A small differential pressure across the piston would result in an insignificant driving force of approximately 10 lb, tending to increase withdraw velocity.

A pressure differential of 295 psi across the collet piston would hold the collet unlatched as long as the driving signal was maintained.

Flow resistance of the exhaust path from the drive would be normal because the ball check valve would be seated at the lower end of its travel by pressure under the drive piston.

4.6.2.3.2.2.7 Ball Check Valve Plug Blows Out

As a means of access for machining the ball check valve cavity, a 1.25-in diameter hole has been drilled in the flange forging. This hole is sealed with a plug of 1.31-in diameter and 0.38-in thickness. A full-penetration weld, utilizing Type 308 stainless steel filler, holds the plug in place. The failure postulated is a circumferential crack in this weld leading to a blowout of the plug.

If the plug were to blow out while the drive was latched, there would be no control rod motion. No pressure differential would exist across the collet piston to unlatch the collet. As in the previous failure, reactor water would flow past the velocity limiter, down the annulus between the drive and thermal sleeve, through the vessel ports and drilled passage, through the ball check valve cage and out the open plug hole to the drywell. The leakage calculations indicate the flow rate would be 350 gpm. This calculation assumes liquid flow, but flashing of the hot reactor water to steam would reduce this rate to a lower value. Drive temperature would rapidly increase and initiate an alarm in the main control room.

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If the plug failure were to occur during control rod withdrawal (it would not be possible to unlatch the drive after such a failure), the collet would relatch at the first locking groove. If the collet were to stick, calculations indicate the control rod withdrawal speed would be 11.8 ft/sec. There would be a large retarding force exerted by the velocity limiter due to a 35 psi pressure differential across the velocity limiter piston. This event requires multiple failures and is therefore beyond the design basis. The Control Rod Drop Analysis (CRDA) assumes a control rod withdrawal speed of 3.11 ft/sec and this speed is bounding for all single failure events.

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4.6.2.3.2.2.8 Drive/Cooling Water Pressure Control Valve Closure (Reactor Pressure, 0 psig)

The pressure to move a drive is generated by the pressure drop of practically the full system flow through the drive/cooling water pressure control valve. This valve is either a motor-operated valve or a standby manual valve; either one is adjusted to a fixed opening. The normal pressure drop across this valve develops a pressure 260 psi in excess of reactor pressure.

If the flow through the drive/cooling water pressure control valve were to be stopped, as by a valve closure or flow blockage, the drive pressure would increase to the shutoff pressure of the supply pump. The occurrence of this condition during withdrawal of a drive at zero vessel pressure will result in a drive pressure increase from 260 psig to no more than 2,000 psig. Calculations indicate that the drive would accelerate from 3 in/sec to approximately 7 in/sec. A pressure differential of 1,970 psi across the collet piston would hold the collet unlatched. Flow would be upward, past the velocity limiter piston, but retarding force would be negligible. Rod movement would stop as soon as the driving signal was removed.

4.6.2.3.2.2.9 Ball Check Valve Fails to Close Passage to Vessel Ports

Should the ball check valve sealing the passage to the vessel ports be dislodged and prevented from reseating following the insert portion of a drive withdrawal sequence, water below the drive piston would return to the reactor through the vessel ports and the annulus between the drive and the housing rather than through the speed control valve. Because the flow resistance of this return path would be lower than normal, the calculated withdrawal speed would be 2 ft/sec. During withdrawal, differential pressure across the collet piston would be approximately 40 psi. Therefore, the collet would tend to latch and would have to stick open before continuous withdrawal at 2 ft/sec could occur. Water would flow upward past the velocity limiter piston, generating a small retarding force of approximately 120 lb.

4.6.2.3.2.2.10 Hydraulic Control Unit Valve Failures

Various failures of the valves in the HCU can be postulated, but none could produce differential pressures approaching those described in the preceding sections and none alone could produce a high velocity withdrawal. Leakage through either one or both of the scram valves produces a pressure that tends to insert the control rod rather than to withdraw it. If the pressure in the scram discharge volume should exceed reactor pressure following a scram, a check valve in the line to the scram discharge header prevents this pressure from operating the drive mechanisms.

4.6.2.3.2.2.11 Collet Fingers Fail to Latch

The failure is presumed to occur when the drive withdraw signal is removed. If the collet fails to latch, the drive continues to with- draw at a fraction of the normal speed. This assumption is made because there is no known means for the collet fingers to become unlocked without some initiating signal. Because the collet fingers will not cam open under a load, accidental application of a down signal does not unlock them. (The drive must be given a short insert signal to unload the fingers and cam them open before the collet can be driven to the unlock position.) If the drive withdrawal valve fails to close following a rod withdrawal, the collet would remain open and the drive continue to move at a reduced speed.

4.6.2.3.2.2.12 Withdrawal Speed Control Valve Failure

Normal withdrawal speed is determined by differential pressures in the drive and is set for a nominal value of 3 in/sec. Withdrawal speed is maintained by the pressure regulating system and is independent of reactor vessel pressure. Tests have shown that accidental opening of the speed control valve to the full-open position produces a velocity of approximately 5 in/sec.

The CRD system prevents unplanned rod withdrawal and it has been shown above that only multiple failures in a drive unit and in its control unit could cause an unplanned rod withdrawal.

4.6.2.3.2.3 Scram Reliability

High scram reliability is the result of a number of features of the CRD system. For example:

1. An individual accumulator is provided for each CRD with sufficient stored energy to scram at any reactor pressure. The reactor vessel itself, at pressures above 600 psi, will supply the necessary force to insert a drive if its accumulator is unavailable.

2. Each drive mechanism has its own scram valves and a dual solenoid scram pilot valve; therefore, only one drive can be affected if a scram valve fails to open. Both pilot valve solenoids must be deenergized to initiate a scram.

3. The RPS and the HCUs are designed so that the scram signal and mode of operation override all others.

4. The collet assembly and index tube are designed so they will not restrain or prevent control rod insertion during scram.

5. The scram discharge volume is monitored for accumulated water and the reactor will scram before the volume is reduced to a point that could interfere with a scram.

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6. The Alternate Rod Insertion (ARI) system is designed such that in the event an RPS scram signal is not received, an independate means is available to automatically vent the scram air header.

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4.6.2.3.2.4 Control Rod Support and Operation

As described in the preceding sections, each control rod is independently supported and controlled as required by safety design bases.

4.6.2.3.3 Control Rod Drive Housing Supports

Downward travel of the CRD housing and its control rod following the postulated housing failure equals the sum of these distances: 1) the compression of the disc springs under dynamic loading, and 2) the initial gap between the grid and the bottom contact surface of the CRD flange. If the reactor were cold and pressurized, the downward motion of the control rod would be limited to the spring compression (approximately 2 in) plus a gap of approximately 1 in. If the reactor were hot and pressurized, the gap would be approximately 3/4 in and the spring compression would be slightly less than in the cold condition. In either case, the control rod movement following a housing failure is substantially limited below one drive "notch" movement (6 in). Sudden withdrawal of any control rod through a distance of one drive notch at any position in the core does not produce a transient sufficient to damage any radioactive material barrier.

The CRD housing supports are in place during power operation and when the nuclear system is pressurized. If a control rod is ejected during shutdown, the reactor remains subcritical because it is designed to remain subcritical with any one control rod fully withdrawn at any time.

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At plant operating temperature, a gap of approximately 3/4 in exists between the CRD housing and the supports. At lower temperatures the gap is greater. Because the supports do not contact any of the CRD housing except during the postulated accident condition, vertical contact stresses are prevented. Inspection and testing of CRD housing supports is discussed in Section 4.6.3.2.

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APPENDIX C

BWROG COMMENTS ON THE DRAFT SAFETY EVALUATION

On June 3, 2004 Robert Vita (Entergy), in representation of the BWROG BPWS committee, placed a call to Bo Pham of the NRC to share the following editorial comments with regard to the NRC's proposed Safety Evaluation (SE) for the Improved Banked Position Withdrawal Sequence (BPWS) Control Rod Insertion Process Licensing Topical Report (LTR):

- In Section 3.0 entitled "Technical Evaluation", first sentence, the phrase "...prevent a CRDA from occurring during startup,..." should read "...mitigate the consequences of a CRDA from occurring during startup,..."
- The first sentence in the fourth paragraph of Section 3.0 entitled "Technical Evaluation", the phrase "...control rod drop (CRD)" should read "...control rod drive (CRD)"

The NRC indicated that they would make the noted changes with reference to the telephone conversation.