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January 17, 1986

Docket No. 50-245
B11951

Office of Nuclear Reactor Regulation
Attn: Mr. C. I. Grimes, Director
Integrated Safety Assessment Project Directorate
Division of PWR Licensing - B
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Gentlemen:

Millstone Nuclear Power Station, Unit No. 1
Environmental Qualification of Electrical Equipment
Request for Exemption

Introduction

On November 20, 1985, the Commission issued a Memorandum and Order to Northeast Nuclear Energy Company (NNECO) regarding the deadline for environmental qualification (EQ) of electrical equipment at Millstone Unit No. 1. The subject of the Memorandum and Order was the Commission's decision regarding eleven (11) motor operators for motor-operated valves used in safety-related systems at Millstone Unit No. 1, for which NNECO had requested a schedular extension beyond November 30, 1985. NNECO had requested an extension beyond November 30, 1985 in order to complete its evaluation of the 11 motor operators in the context of the Integrated Safety Assessment Program (ISAP). In deciding this issue, the Commission had the benefit of NNECO's submittals of September 30, 1985(1) and October 29, 1985,(2) the Staff's analysis of the issue as presented in SECY-85-345, and presentations by the Staff and by NNECO to the Commission on November 5, 1985. In its November 20, 1985 Memorandum and Order, the Commission approved an extension of the deadline for EQ for Millstone Unit No. 1 to the next outage of sufficient duration after the Staff has made a determination on whether an exemption to Section 50.49 can be granted, or to the next refueling outage, but in no event later than August 30, 1987.

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(1) J. F. Opeka letter to Nunzio J. Palladino, dated September 30, 1985, Environmental Qualification of Electrical Equipment, Schedular Extension Requests.

(2) J. F. Opeka letter to Nunzio J. Palladino, dated October 29, 1985, Environmental Qualification of Electrical Equipment, Schedular Extension Requests.

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In support of the request for an extension of the equipment qualification deadline beyond November 30, 1985 for the 11 valve motor operators, it was made clear that:

- o NNECO has in good faith diligently pursued qualification of the valve operators at issue,

- o there is no immediate safety concern regarding the qualification of these valve motor operators, and

- o the Commission's ISAP Policy Statement (49 Fed. Reg. 45113, November 15, 1984) endorses integration of actions such as the qualification of the 11 valve motor operators into the ISAP.

Additionally, as discussed in NNECO's submittals and in SECY-85-345, and as noted during the November 5, 1985 Commission meeting, based upon preliminary ISAP evaluations of the valve operators and the justifications for continued operation provided by NNECO(3) it appeared as though permanent exemptions from the requirements of 10CFR50.49 might be warranted.

NNECO has completed its evaluation of the 11 remaining valve motor operators which are currently included on the EQ Master List and for which full EQ has not been demonstrated, and determined that none of these 11 valve operators warrants replacement. Accordingly, NNECO hereby requests an exemption, pursuant to 10CFR50.12, from the requirements of 10CFR50.49 for these components. The bases for the exemption request are provided in this letter and its attachment.

Exemption Criteria

The Commission's regulations, specifically 10CFR50.12(a), provide that exemptions may be granted from the regulations in 10CFR Part 50 provided that they "are authorized by law and will not endanger life or property or the common defense and security and are otherwise in the public interest."

On December 12, 1985, the Commission published the final revisions to 10CFR50.12(a), regarding standards to be applied in granting exemptions (50 Fed. Reg. 50764, December 12, 1985). The purpose of the final rule was to revise and clarify the criteria for granting of exemptions. The revised 10CFR50.12(a), which became effective on January 13, 1986, reads as follows:

Section 50.12 Specific Exemptions

(a) The Commission may, upon application by any interested person or upon its own initiative grant exemptions from the requirements of the regulations of this part, which are:

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(3) See, for example, NNECO submittals of September 30, 1985 (regarding the schedular extension request), October 29, 1985 (providing updated justifications for continued operation) and October 17, 1985 (providing assessments of public safety impact of all remaining Millstone Unit No. 1 EQ modifications).

(1) Authorized by law, will not present an undue risk to the public health and safety, and are consistent with the common defense and security.

(2) The Commission will not consider granting an exemption unless special circumstances are present. Special circumstances are present whenever:

(i) Application of the regulation in the particular circumstances would be in conflict with other rules or requirements of the Commission; or

(ii) Application of the regulation in the particular circumstances would not serve the underlying purpose of the rule or is not necessary to achieve the underlying purpose of the rule; or

(iii) Compliance would result in undue hardship or other costs that are significantly in excess of those contemplated when the regulation was adopted or that are significantly in excess of those incurred by others similarly situated; or

(iv) The exemption would result in an overall benefit to the public health and safety that compensates for any decrease in safety that may result from the grant of the exemption; or

(v) The exemption would provide only temporary relief from the application regulation and the licensee or applicant had made good faith efforts to comply with the regulation;

(vi) There is present any other material circumstance not considered when the regulation was adopted for which it would be in the public interest to grant an exemption. If such condition is relied on exclusively for satisfying paragraph (a)(2) of this section, the exemption shall not be granted until the Executive Director for Operations has consulted with the Commission.

Our request for exemption is based upon the revised 10CFR50.12(a), above. An evaluation of each of the 11 motor operators is presented in Attachment 1. The information in Attachment 1 includes identification of each valve motor operator, the system affected, a brief statement of the safety function of each valve, a description of how and under what conditions operation of the valves might be required, and an evaluation of the safety impact of an unqualified operator and the benefit that could be realized if the operator were replaced with fully qualified equipment. Based on this information, NNECO has concluded that an exemption from the requirements of 10CFR50.49 for the subject 11 motor operators is justified pursuant to paragraph 50.12(a)(2)(ii) of the final rule. That is, it is:

- Authorized by law,
- Will not present an undue risk to public health and safety and
- Is consistent with the common defense and security.

Additionally, as the information contained in the attachment to this letter will demonstrate:

o Application of the regulation in the particular circumstances would not serve the underlying purpose of the rule or is not necessary to achieve the underlying purpose of the rule (10CFR50.12(a)(2)(ii)).

In general, the intent of the Commission's regulations and other NRC requirements is to provide reasonable assurance that operation of nuclear power plants does not pose an undue risk to the health and safety of the public. The intent of specific regulations such as 10CFR50.49 (EQ) is to set standards which will provide reasonable assurance that the individual contributions to risk posed by specific issues or concerns will be low, such that overall plant risk remains acceptably low. While compliance with the regulations will, for the most part, provide reasonable assurance that plant operation does not pose undue risk to the public, noncompliance does not necessarily represent an unacceptable risk.

In the specific case of the Commission's requirements for EQ of electrical equipment as contained in 10CFR50.49, the underlying purpose of these requirements is to provide reasonable assurance that the risk posed by the potential for failure of electrical equipment due to harsh environmental conditions is low. This is not to say that this risk contribution must be totally eliminated, as that would be inconsistent with the intent of the regulations in general, and most likely would be impossible to achieve. For example, the Commission's requirements concerning seismic design dictate that a nuclear power plant be capable of withstanding the effects of a certain level of ground motion which has an extremely low (but not zero) probability of being exceeded. It is recognized that more severe levels of ground motion could occur; however, the chance of occurrence is judged to be extremely low and thus the risk posed from such a low probability event is deemed acceptable. Therefore, in the case of EQ, satisfying the underlying purpose of the rule does not require that it be unequivocably demonstrated from a deterministic standpoint that potential equipment failures due to harsh environmental conditions will pose absolutely no risk, although in several cases this is essentially the case. Rather, a showing that the risk posed by unqualified equipment is an acceptably low contribution to overall plant risk is an adequate demonstration that the underlying purpose of the regulation is satisfied.

One means for assessing the contribution to risk posed by individual issues is through the use of probabilistic risk assessment techniques. In accordance with the ISAP program plan,[4] NNECO submitted a plant-specific Probabilistic Safety Study (PSS) for Millstone Unit No. 1.[5] Insights from this study have been applied in our evaluation of the 11 motor operators which are the subject of this exemption request.

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(4) SECY-84-133, Integrated Safety Assessment Program, dated March 23, 1984.

(5) J. F. Opeka letter to J. A. Zwolinski, dated July 10, 1985, Probabilistic Safety Study Results and Summary Report.

Attachment 1 contains evaluations for each of the 11 motor operators for which compliance with EQ requirements has not been demonstrated. Our primary method for evaluating the safety significance of the motor operators is a deterministic, defense-in-depth evaluation of safety impact. In most cases, it is demonstrated from a deterministic standpoint that replacement of the motor operator with a fully qualified operator will have no impact on plant safety or public risk. For other valve operators where it could not be shown conclusively that replacement would have no safety impact, the PSS results were utilized to quantify the benefits that would result from replacement. These evaluations demonstrate that there would either be no benefit in replacing the operators, or that the benefit that would result would be extremely small. Thus, it is clear that implementation of these modifications is not necessary to serve the underlying purpose of the rule and we conclude that the requested exemptions are justified and appropriate.

In our evaluations of individual motor operators, we have utilized insights from the PSS to calculate a predicted benefit due to environmental qualification which is based on the change in core melt frequency. We believe that the change in core melt frequency is an appropriate standard to use in evaluating the benefits of valve operator replacement. While we recognize that there are events which could occur at a nuclear power plant that would be considered significant but which would not result in a core melt, these events would pose little or no actual public risk. The only events which pose a significant risk to the public are those which result in severe core damage or core melt and a major loss of containment integrity. Additionally, as shown in the evaluations presented in Attachment 1, for most of the subject valve operators it can be demonstrated that replacement with qualified operators will have zero impact on safety. For example, in several cases the environment for which the operator is not qualified can result only if the system in which the valve is located has already failed, and thus having a fully qualified operator is moot. Additional discussion regarding consideration of non-core melt accidents and the appropriateness of using change in core melt frequency as a standard was provided in a letter from J. F. Opeka to N. J. Palladino, dated January 6, 1986.

Although this exemption request is based primarily on criterion 50.12(a)(2)(ii), above, there is a reasonable basis for application of 50.12(a)(2)(iii) in support of this request. To the best of our knowledge, Crane-Teledyne motor operators have not been utilized in similar applications in any other U.S. nuclear power plant. Most licensees have been able to demonstrate qualification of existing valve motor operators without extensive wholesale replacements being required, whereas our efforts at demonstrating qualification of these Crane-Teledyne operators have been unsuccessful. Additionally, the replacement operators were larger than the original operators, and were not directly compatible with the installed valves. For example, new valve yokes have been required for some valves, and in many cases the mounting configurations have been changed significantly. Also, piping geometry in one case is such that the new operator cannot be installed unless the piping is cut and the valve physically rotated to provide adequate clearance. Finally, in every case, replacement of the valve operator necessitates an evaluation of the impact on seismic qualification of the piping system. We believe that all of these factors combined with the expected radiological exposures associated with the replacements (discussed below), are in excess of those contemplated when the regulation was adopted and in excess of those typically incurred by other licensees.

Although not directly relevant to the evaluation of this exemption request, as part of the implementation of the ISAP concept within Northeast Utilities, criteria have been developed by which other potential impacts of proposed design changes can be evaluated. These criteria include the impact on personnel safety (radiological and industrial safety), personnel productivity and plant economic performance. We have examined the impact of the proposed valve operator replacements with respect to these attributes and concluded that, with the exception of radiological exposure and some personnel hazards (e.g., use of high scaffolding inside the drywell) associated with installation, there would be no impact, either positive or negative, in any of these areas. This is due to the fact that the modifications are essentially one-for-one equipment replacements, and thus the costs, manpower, risks, and frequency of surveillance and testing, etc., would not change. Additionally, since replacement with qualified equipment only affects operability of the valves in a post-accident (harsh environment) scenario, there is zero impact on plant reliability or availability (i.e., plant economic performance).

While this exemption request concerns only the 11 remaining operators, our evaluations for these other attributes considered all twenty-eight (28) MOVs discussed in our September 30, 1985 letter. In one case (valve IC-3) a benefit was predicted in the economic performance attribute. Based on historical failures of this valve and the Technical Specification limit on power level (40%) with the isolation condenser inoperable, an availability improvement of approximately 1 hour of full power generation per year was predicted. We note that as committed in our September 30, 1985 letter, the operator for valve IC-3 was replaced with a fully qualified operator during the 1985 refueling outage.

The issue of occupational radiological exposure during installation is the only aspect other than safety which is relevant to this case. As noted by NNECO during the November 5, 1985 Commission meeting on this subject, it is expected that approximately 165 man-rem of exposure would be required to install the 11 motor operators. While we firmly believe that our deterministic evaluations, as supported by the probabilistic findings, form an adequate basis for granting the requested exemption, this conclusion becomes even more clear given the significant exposure that would result from installation of the motor operators. We view this predicted exposure as a significant factor in determining the appropriateness of these modifications. Due in part to higher than average cumulative occupational radiological exposures at our nuclear facilities, ALARA considerations have been given a high degree of consideration in our outage planning and during routine operation. Corporate goals have been established to achieve reductions in exposure levels, and in order to achieve success in this area we are elevating the importance of ALARA considerations in all of our backfit planning. We note that this projected exposure is more than a factor of 2 greater than the maximum benefit (in terms of averted population dose) that could possibly be achieved through replacement of the operators. Sound engineering judgement would not support the actual exposure of approximately 165 man-rem to avert a hypothetical population dose of less than half of that amount. Again, although this is not a direct consideration in our exemption request, we believe that it provides additional support for granting the exemption.

Conclusion

In summary, NNECO has concluded that replacement of the 11 motor operators identified in Attachment 1 to satisfy EQ requirements is not justified, and that

an exemption for these operators from the requirements of 10CFR50.49 should be granted, pursuant to 10CFR50.12(a)(2)(ii), which becomes effective on January 13, 1986. Upon granting of the requested exemption, Millstone Unit No. 1 will be in full compliance with the provisions of 10CFR50.49, and all required modifications will have been completed prior to start-up from the most recent refueling outage, which began on October 26, 1985. Thus, all necessary modifications would have been completed in accordance with the schedule dictated by the Commission in Generic Letter 85-15, dated August 6, 1985.

Finally, should the Staff agree that exemptions for these motor operators are appropriate, we request that the November 20, 1985 Memorandum and Order be revised or rescinded, as necessary.

In accordance with the requirements of 10CFR170.21, enclosed is the application fee of $150.00.

As always, should you have any questions concerning this request, please contact us.

Very truly yours,

NORTHEAST NUCLEAR ENERGY COMPANY

J. F. Opeka

J. F. Opeka
Senior Vice President

Docket No. 50-245

Attachment 1

Millstone Nuclear Power Station, Unit No. 1

Environmental Qualification of Electrical Equipment

Request for Exemption

January, 1986

Valves: RR-2A
RR-2B

System: Recirculation

Function: Recirculation Pump Discharge Isolation Valves

Description: Valves RR-2A and 2B are the reactor recirculation pump discharge isolation valves, and are normally open.

In the event of a LOCA, the Low Pressure Coolant Injection (LPCI) loop select logic will detect which recirculation loop ("A" or "B" loop) is broken, and will open and close valves as required to ensure that all LPCI injection flow is directed into the intact loop and to the core. The LPCI system serves to reflood the reactor vessel to two-thirds of the active core height following a LOCA.

As shown on Figure 1, attached, the LPCI system injects into the recirculation loops between the recirculation pump discharge valves and the reactor vessel. In the event of a LOCA, however, LPCI will inject into only one of the two recirculation loops, as the loop select logic will always choose only one loop. The loop select logic will determine the broken loop, and will send a "close" signal to the recirculation pump discharge valve in the intact loop (Valve RR-2A will close if the break is in loop "B", while valve RR-2B will close if the break is in loop "A"). The recirculation pump discharge valve in the broken loop does not receive a "close" signal, and remains in its normally open position. This results in a rapid blowdown of the reactor vessel permitting earlier low pressure injection than would otherwise be the case if this valve closed, and ensures that the vessel remains depressurized during reflood. The LPCI loop select logic also commands the normally closed injection valve in the broken loop (LP-10A or 10B) to remain closed and the injection valve in the intact loop to open. As a result of modifications performed during the 1985 refueling outage, valves LP-10A and 10B fully meet the requirements of 10CFR50.49.

Evaluation:

Proper functioning of the LPCI system requires only RR-2A or 2B to close, but not both. If the discharge isolation valve in the intact loop fails to close, as directed by the loop selected logic, the LPCI flow injected into that loop could flow back through the recirculation pump and into the annular region between the core shroud and the reactor vessel. This flow could then pass around the core shroud, bypassing the core, and out through the break in the other recirculation loop. This would negate the effectiveness of the LPCI system for reflooding the reactor vessel.

It is estimated that flow bypass of the core is a concern only for breaks greater than 0.01 ft^2. For smaller breaks, vessel water level can be recovered even if spillage of LPCI flow through the break occurs. Thus, proper closure of a recirculation pump discharge valve is required for LPCI system success for only larger break sizes.

It is important to note that although the LPCI system is designed as a 2 train system, the entire LPCI system can be disabled by a single active failure. As stated above, in the event of a LOCA, the loop select logic will always select only one loop for injection. Should the injection valve (LP-10A or 10B) in the chosen (intact) loop fail to open, the entire LPCI capability is lost. For this reason, the design basis large break LOCA analysis for Millstone Unit No. 1 does not take any credit for functioning of the LPCI system as it has been determined that, from a single failure standpoint, failure of the LPCI injection valve to open is more limiting than failure of an emergency power source (e.g. - diesel or gas turbine generator). In this case, the core spray system is fully capable of providing adequate core cooling. We note that as a result of modifications performed during the 1985 refueling outage, the core spray system is fully qualified to 10CFR50.49 requirements.

In an October 17, 1985 letter from J. F. Opeka to C. I. Grimes, the Staff was provided with the results of our evaluation of RR-2A and 2B utilizing the PSS. This evaluation concluded that replacement of these two valve operators would result in a decrease in core melt frequency of approximately 1.5×10^{-6}/year. This is a conservative estimate of the maximum possible benefit which could be obtained, since it assumed that without fully qualified motor operators for RR-2A and 2B, the probability of failure of LPCI is 1.0, even though there is some probability that the valve which is required to close (2A or 2B) will function properly even without full qualification documentation. This change in core melt frequency corresponds to a reduction of approximately 60 man-rem over the remaining life of the plant.

The LPCI system can also be used in the Alternate Shutdown Cooling/Containment cooling mode, as described in the Millstone Unit No. 1 PSS. In this mode of operation, operation of RR-2A or 2B is not required for system success. Thus, replacement of the motor operators for RR-2A and 2B will have no impact on the reliability of Alternate Shutdown Cooling/Containment Cooling.

The failure of these valves to function properly will not degrade other safety functions. It is conceivable that valve position indication in the control room could be incorrect. However, operator actions are based on symptom-oriented Emergency Operating Procedures (EOPs) which focus operator attention on restoring and maintaining reactor water level. Should a recirculation pump discharge isolation valve be "closed" yet indicate "open" due to a harsh environment, there would not be any additional impact on vessel inventory. Hence, no actions by the operator beyond those necessitated by the original event symptoms would be warranted. Should a recirculation pump discharge isolation valve be "open" yet indicate "closed", a diversion of LPCI injection flow would exist. Again, the operator is not directed to identify and correct the diversion, but is merely directed to start additional pumps and provide enough makeup to restore and maintain vessel water level. Since the operator does not rely on these valve position indications, inaccurate valve position indications will not mislead the operator.

Valves: IC-2
IC-4

System: Isolation Condenser

Function: Primary Containment Isolation for an Isolation Condenser Line Break

Description:

Isolation valves IC-1, IC-2, IC-3, and IC-4 for the Isolation Condenser steam and condensate lines are arranged inside and outside of containment as shown in Figure 2. Valves IC-1 and IC-2 are the inboard and outboard steam line isolation valves, respectively while IC-3 and IC-4 are outboard and inboard condensate line isolation valves, respectively. Valves IC-1, IC-2 and IC-4 are normally open, whereas IC-3 is normally closed. Opening of IC-3 is the only action necessary to initiate Isolation Condenser cooling. All four isolation valves receive a "close" signal on a Group IV isolation. A Group IV isolation is initiated by high flow in the Isolation Condenser System, a condition indicative of a pipe break in the Isolation Condenser System.

Evaluation:

Valve IC-2 is located in the reactor building (outside primary containment) on the inlet piping for the Isolation Condenser System. This valve is normally open and will close in the event of a Group IV isolation. If a break occurs in this pipe inside the drywell, closure of IC-2 will not isolate the break. If the break occurs in this line outside the drywell, valve IC-1 (which is inside the drywell and thus not subject to the harsh environment and is additionally fully qualified to 10CFR50.49 requirements) will close and isolate the break. Thus, the only benefit of environmentally qualifying valve IC-2 would be for the case of an Isolation Condenser steam line break outside the drywell combined with a random single failure of valve IC-1.

In an October 17, 1985 letter from J. F. Opeka to C. I. Grimes, the Staff was provided with the results of the probabilistic evaluation of this postulated scenario. The calculated change in core melt frequency was determined to be approximately 2.0×10^{-9}/year, which corresponds to a reduction in risk of approximately 0.1 man rem over the remaining life of the plant.

Valve IC-4 is inside the drywell on the condensate return line from the Isolation Condenser to the "B" recirculation loop. This valve is normally open and closes automatically on a Group IV isolation. In the event of a break in the Isolation Condenser condensate return piping inside of the drywell, closure of IC-4 will not isolate the break. Valves IC-1 (fully qualified) and IC-2 (not exposed to harsh environment) will close to provide containment isolation, and valve IC-3 (fully qualified) will remain closed to provide isolation of the condensate return line. If the break occurs outside the drywell, IC-3 is normally closed and will remain

closed. It is noted that as a result of modifications performed during the 1985 refueling outage, valve IC-3 is now fully qualified to 10CFR50.49 requirements. Also, IC-1 (which would not be in the harsh environment and is additionally fully environmentally qualified) and IC-4 (not exposed to harsh environment) will close to provide containment isolation. Based on the above, it is clear that replacement of the valve operator for IC-4 with a fully environmentally qualified operator will have no impact on the frequency of core melt at Millstone Unit 1.

During normal operation, valves IC-1, IC-2, IC-4 are open and only IC-3 (normally closed) needs to open to initiate Isolation Condenser cooling when required. Therefore, the Isolation Condenser system can perform its intended safety function after a small-break LOCA with fully qualified equipment. IC-2 and IC-4 are normally open and stay open during a LOCA. Any electrical failure which could cause them to change their position is not credible.

The only credible failure mode for these valves once they have completed their safety function (i.e., the valves close following an Isolation Condenser line break) is one where the motor operator continues to operate against the seat of the valve until the operator stops due to motor failure or protective action. This type of failure would not preclude the fulfillment of the safety function of these valves or result in the degradation of any function. Even in the very unlikely case where either of these valves fail to close due to a harsh environment, no safety function would be degraded since other environmentally qualified containment isolation valves in the Isolation Condenser System would perform the safety function. The only exception would be for the case of a break outside the drywell combined with an assumed single failure of IC-1. However, as discussed earlier, this is an extremely low probability event.

It is conceivable that valve position indication in the control room could be incorrect. However, operator actions are based on symptom-oriented Emergency Operating Procedures (EOPs) which focus operator attention on restoring and maintaining reactor water level, controlling the RPV pressure, and cooling the containment. Should one of these valves be "closed" yet indicate "open" due to a harsh environment, there would not be any additional impact on vessel inventory or containment cooling status. Hence, no actions by the operator beyond those necessitated by the original event symptoms would be necessary to accomplish the intended safety function. Should one of these valves be "open" yet indicate "closed" due to the harsh environment, the safety function (i.e., isolation) would be accomplished by the redundant valve. The symptom-oriented EOPs would continue to focus operator attention on the critical functions of vessel inventory control and containment cooling. Since the operator does not rely on these valve position indications, inaccurate valve position indications will not mislead the operator.

Valves: LP-15A
LP-15B
LP-16A
LP-16B

System: Low Pressure Coolant Injection

Function: To provide drywell spray

Description:

Valves LP-15A and B, and LP-16A and B are arranged in the Low Pressure Coolant Injection (LPCI) system as depicted in Figure 3. In the event of a large break LOCA, or a small break LOCA in conjunction with Automatic Pressure Relief (APR), the LPCI system provides emergency core cooling injection and long-term recirculation.

An additional function of LPCI is the LPCI/containment cooling mode of operation. This is effected through the use of drywell spray or torus cooling, and is achieved by recirculating LPCI flow via the emergency cooling water heat exchangers, which are in turn cooled by the emergency service water system. LP-15A and B and LP-16A and B are AC and DC powered valves, respectively, which could be opened post-accident to provide drywell spray flow. In this mode of operation, the LPCI pumps can take suction from either the ECCS suction header on the suppression pool (normal) or the condensate storage tank (alternate). After leaving the LPCI pumps, water is directed either through the LPCI heat exchangers or around them through bypass valves. Water is then directed through the drywell spray valves (LP-15 and LP-16) to one of two eight inch spray ring headers inside the drywell. Both spray ring headers can be used simultaneously.

Evaluation:

Valves LP-15A and B and LP-16A and B are located in the reactor building and are normally closed. Operator action is required to open these valves and initiate drywell spray.

In the event of a break in the reactor building, these motor operators would be exposed to a harsh environment. However, a break outside primary containment does not require use of the drywell spray, and therefore these valves would not be used. In the event of a break inside the drywell, the motor operators would not be exposed to the harsh environment since they are located outside containment. In this scenario, the valve operators could be subjected to high radiation from recirculated LPCI water if gross fuel failure occurs. However, if the LPCI system (or core spray system, which is fully qualified) is operating, gross fuel failure is not expected. Gross fuel failure could occur only if LPCI is unavailable. Therefore, since the LPCI pumps also provide the drywell spray flow, qualifying these valves for conditions which can result only if LPCI has failed will have no impact on the frequency of a core melt or core damage accident.

Functioning of the drywell spray has not been assigned in the design basis calculations of post-accident containment conditions. These analyses demonstrate that even without any drywell spray cooling, the post-accident pressure and temperature conditions would not result in loss of the containment function. As such, these valves serve no essentia' safe.y function following a design basis accident. If no manual action is taken by the operator to open these valves, no other safety function would be degraded.

These four valves, which are all located outside containment, are not required to operate during a HELB outside containment. The Emergency Operating Procedures (EOPs) would only direct the operator to manually open these valves when drywell pressure is greater than 15 psig (i.e., break inside containment). For a LOCA, the only deficiencies in the qualification documentation concern radiation and aging considerations. The deleterious effects resulting from these parameters are time dependent phenomena which could not reasonably be expected to result in equipment failure in the short term. In addition, no fuel damage is expected if LPCI is operating. Therefore, these valves would not be exposed to high radiation levels. If LPCI pumps are not operating, operation of these valves is irrelevant. These valves are normally closed. Therefore, unless the operator opens these valves, they continue to provide the containment isolation function. As such, if these valves are opened by the operators, we fully expect them to close after completion of drywell spray. Therefore, no other safety function would be degraded. There is no credible failure mode where these valves could change state inadvertently (i.e. - open without operator action). Additional discussion regarding the use of the drywell spray was provided in a January 6, 1986 letter from J. F. Opeka to N. J. Palladino.

It is conceivable that valve position indication in the control room could be incorrect. However, operator actions are based on symptom-oriented Emergency Operating Procedures (EOPs) which focus operator attention on restoring and maintaining reactor water level and containment temperature and pressure. Should a drywell spray valve be "closed" yet indicate "open" due to a harsh environment, there would not be any additional impact on reactor vessel inventory or containment cooling status. Also, the operator would easily be able to infer from the drywell pressure transient that the sprays are not on. Hence, no actions by the operator beyond those necessitated by the original event symptoms would be warranted. Should a drywell spray valve be "open" yet indicate "closed", a diversion of LPCI injection flow would exist, however, the operator will know that the sprays are on as the drywell pressure would be decreasing rapidly. Again, the operator is not directed to identify and correct the diversion, but is merely directed to start additional pumps and provide enough makeup to restore and maintain vessel water level. Since the operator does not rely on these valve position indications, inaccurate valve position indications will not mislead the operator.

Valves: CU-2
CU-3

System: Reactor Water Cleanup (RWCU)

Function: RWCU Primary Containment Isolation Valves

Description:

Valves CU-2 and CU-3 are isolation valves for the Reactor Water Clean-up (RWCU) system. These valves are located inside and outside of primary containment (drywell) as depicted on Figure 4. Lines that penetrate the reactor coolant pressure boundary and primary containment, interface with auxiliary systems that are not required during isolation conditions, and are located outside of the primary containment are isolated on a Group III isolation signal. The RWCU system is one system affected by this isolation. The Group III isolation senses reactor low water level and initiates an isolation to limit the loss of coolant inventory in the event the leak is from a Group III system. Since the RWCU is a closed system, its isolation will limit the loss of coolant inventory only if the leak is from the RWCU system.

Evaluation:

RWCU isolation valves CU-2 and CU-3 are normally open and located in series in the RWCU piping. These valves get a signal to close if high flow is detected in the RWCU piping or upon low reactor water level. The principal cause of high flow would be due to a break in the RWCU piping. Thus, the only potential benefit in qualifying these valves would be to isolate a break in the RWCU piping outside of the drywell.

If a break occurs inside the drywell, closing of CU-2 or CU-3 will have no impact on break isolation. Additionally, CU-3 is located outside the drywell and would not be exposed to a harsh environment, and would be able to provide the containment isolation function. If the break occurs outside the drywell, CU-2 will not be exposed to the harsh environment and will isolate the break. The only benefit in environmentally qualifying CU-3 is in the scenario with a break in the RWCU piping outside of containment and a random failure of CU-2 to close. In that case, closure of CU-3 would allow for isolation of the break.

In Reference (1), the Staff was provided with the results of a probabilistic evaluation of this scenario. The calculated change in core melt frequency due to qualification of CU-3 was determined to be approximately 2.5×10^{-7}/year, which corresponds to a reduction in risk of approximately 18 man-rem over the remaining life of the plant. There is no projected change in core melt frequency due to qualification of CU-2.

The only deficiency in the qualification documentation concerns radiation and aging considerations. The deleterious effects resulting from these parameters are time-dependent phenomena which could not reasonably be expected to result in equipment failure in the short term. This equipment will have completed its safety function within the first 30 seconds of an incident. Based upon our

engineering evaluation (considering the equipment operating time and expected effects associated with the qualification discrepancy), partial test data of similar valves, and information provided by the manufacturer, we have a high level of confidence that these valves would satisfactorily perform their design function.

The only credible failure mode for these valves once they have completed their safety function (i.e., the valves close) is one where the motor operator continues to operate against the seat of the valve until the operator stops due to motor failure or protective action. This type of failure would not preclude the fulfillment of the safety function of these valves or result in the degradation of any safety function. Even in the very unlikely case where CU-2 (or CU-3) fails to close due to a harsh environment, no safety function would be degraded since CU-3 (or CU-2), which would not be located in the harsh environment, would perform the safety function.

It is conceivable that valve position indication in the control room could be incorrect. However, operator actions are based on symptom-oriented Emergency Operating Procedures (EOPs) which focus operator attention on restoring and maintaining reactor water level and cooling the containment. Should one of these valves be "closed" yet indicate "open" due to a harsh environment, there would not be any additional impact on vessel inventory or containment cooling status. Hence, no actions by the operator beyond those necessitated by the original event symptoms would be necessary to accomplish the intended safety function. Should one of these valves be "open" yet indicate "closed" due to the harsh environment, the safety function (i.e., isolation) would be accomplished by the redundant valve. The symptom-oriented EOPs would continue to focus operator attention on the critical functions of vessel inventory control and containment cooling. Since the operator does not rely on these valve position indications, inaccurate valve position indications will not mislead the operator.

Valve: MW-96A

System: Condensate Transfer

Function: Emergency Condensate Transfer Pump Discharge Stop Valve

Description:

The Feedwater Coolant Injection System (FWCI)[1] takes suction from the main condenser hotwell and pumps water at high pressure into the reactor vessel via the feedwater lines. In cases where the main steam system and/or main condenser are not available, makeup to the condenser hotwell is necessary in order to maintain adequate net positive suction head for the condensate pumps (part of the FWCI system). Makeup to the hotwell is provided by the emergency condensate transfer pump from the Condensate Storage Tank (CST).

Evaluation:

Valve MW-96A lacks qualification documentation for aging and for radiation exposure. MW-96A is located in the vicinity of the Reactor Building Equipment Drain Tank (RBEDT) which could become radioactive only in the very unlikely event of an accident which results in fuel damage. Since this valve would not be located in a harsh steam or temperature environment, the qualification effort consists of adding shielding between the valve and the RBEDT. Aging concerns are addressed through routine maintenance and testing.

For any transient or LOCA at Millstone Unit 1, no fuel damage is expected to occur if either feedwater or FWCI operates successfully; fuel damage can occur only if this system has failed or is ineffective. Thus, the RBEDT could become radioactive only if the FWCI system has failed or is ineffective. The additional shielding would ensure operability of MW-96A only for scenarios in which it is not required (i.e. - FWCI has failed). MW-96A has no other safety function. Therefore, adding a radiation shield for MW-96A to meet radiation qualification requirements would result in no safety benefit. Our preliminary estimate indicates that an exposure of approximately 4 man-rem would be incurred during implementation of this modification.

Based upon the above information, as well as partial test data for similar valves and information provided by the manufacturer, we have a high level of confidence that this valve will function properly. The only credible failure mode for this valve once it has completed its safety function (i.e., the valve opens) is one where the motor operator continues to operate against the backseat until the operator stops due to motor failure or protective action. This type of failure would not preclude the fulfillment of the safety function of this valve or result in the degradation of any safety function.

(1) The FWCI system at Millstone Unit 1 is a high pressure make-up system which utilizes one train of the normal feedwater system, and is comparable to a High Pressure Coolant Injection or High Pressure Core Spray System.

The operator will be concerned with the status of valve MW-96A only if FWCI is being used to maintain the RPV level. In these scenarios, there is no conceivable way in which the RBEDT could become radioactive. Therefore, the valve position indication should function properly for the scenario in which it is required. Additionally, the operator can infer valve position from hot well level indication.

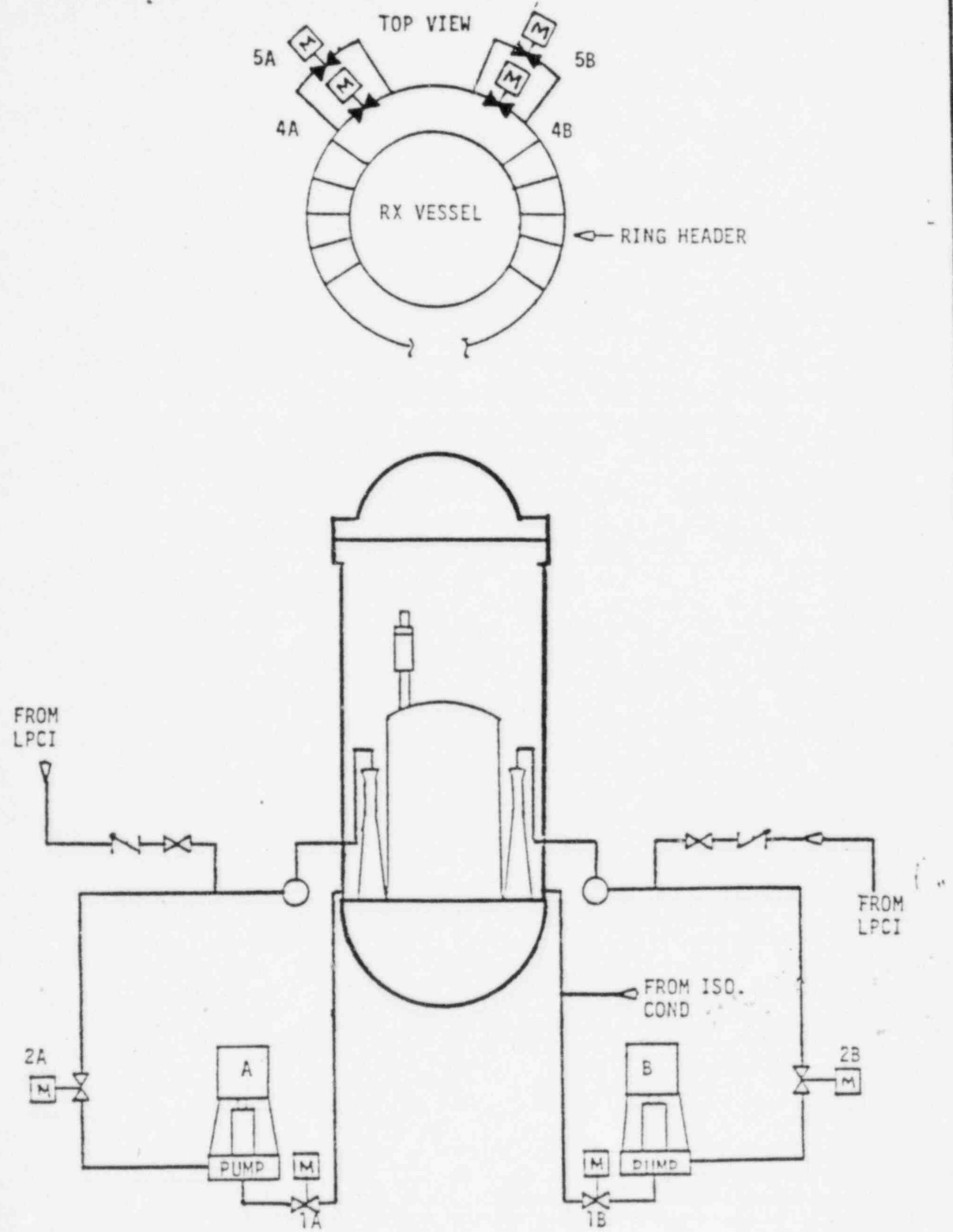


FIGURE 1

RECIRCULATION SYSTEM

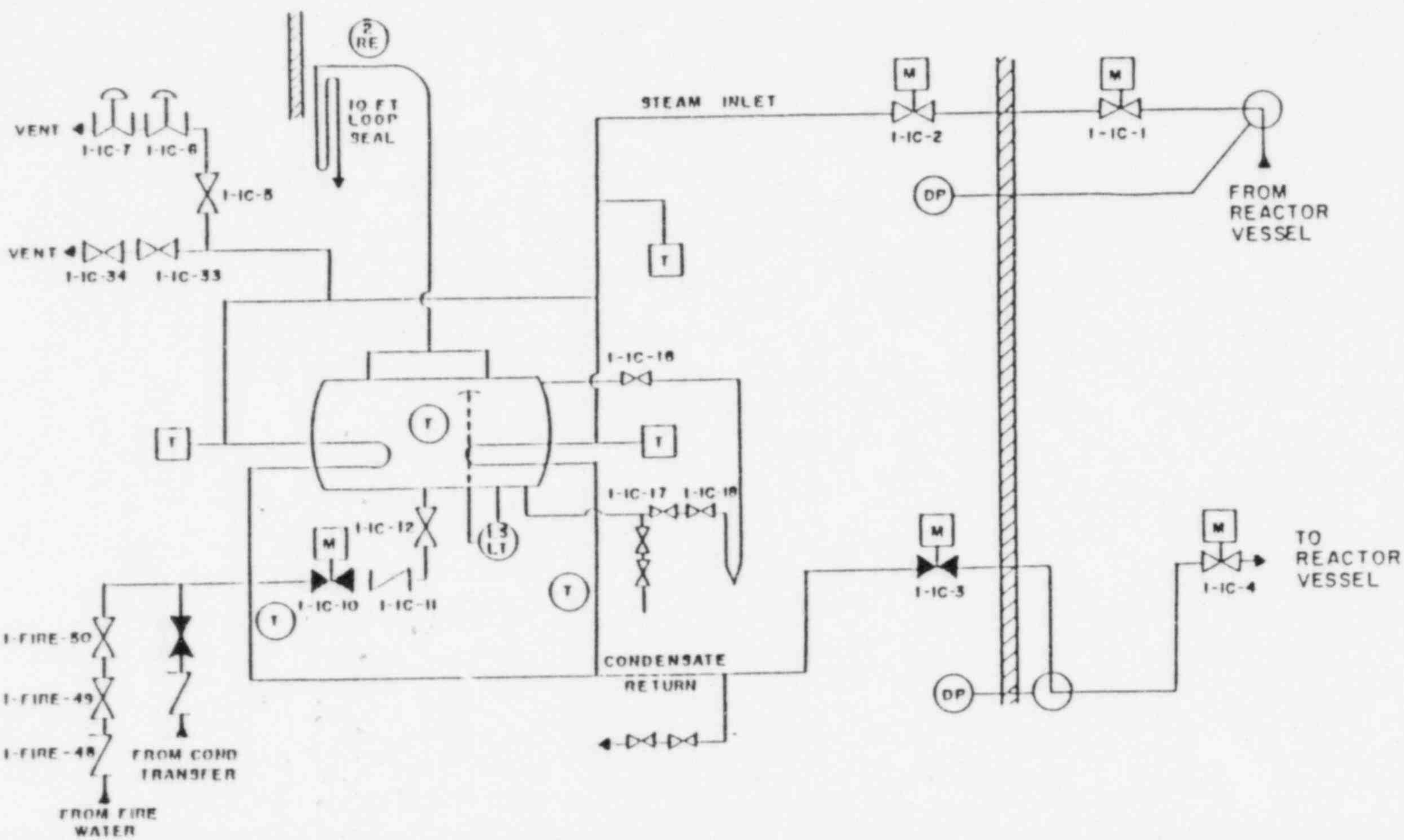
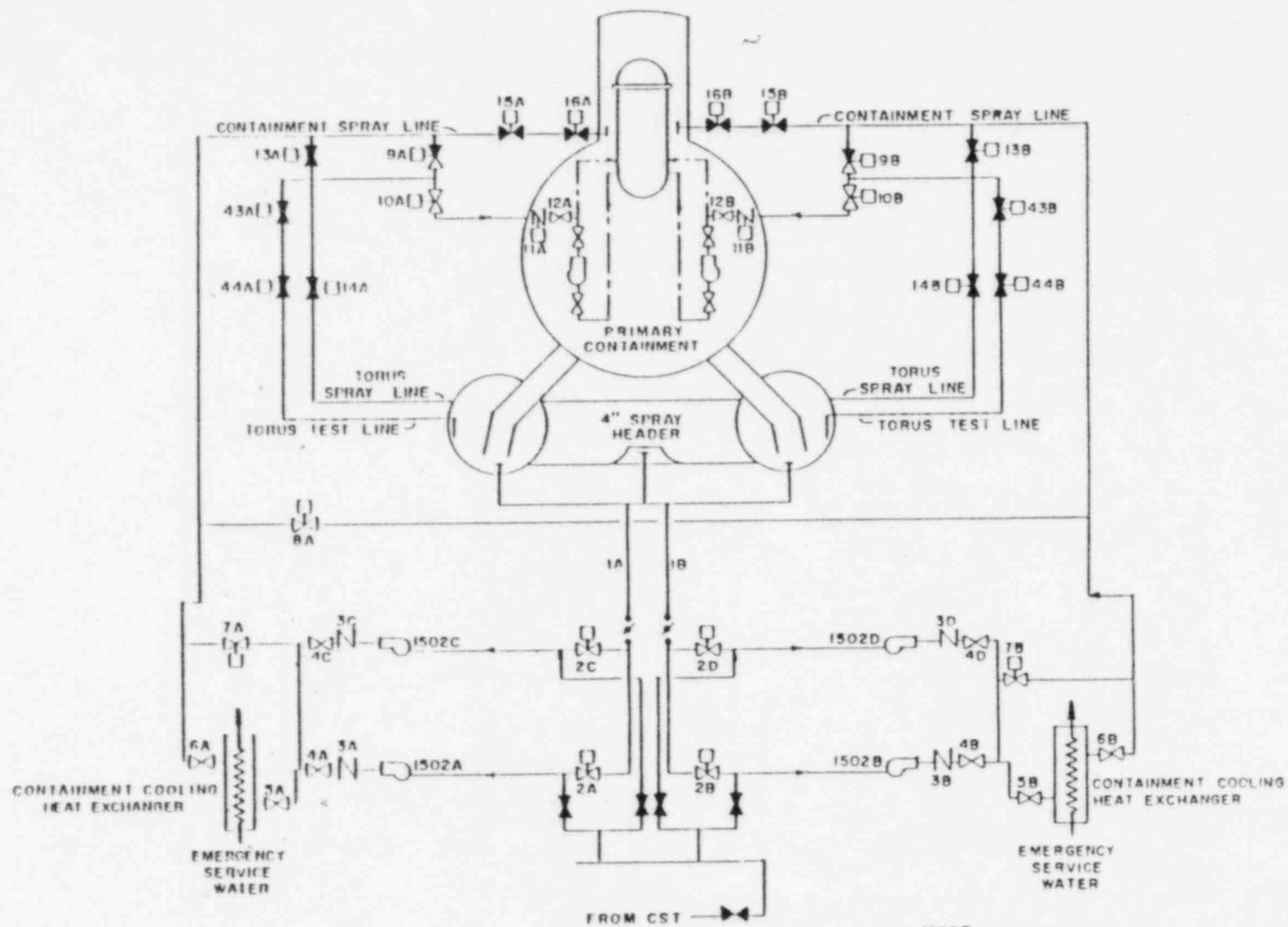


FIGURE 2 ISOLATION CONDENSER SYSTEM DIAGRAM



NOTE:

ALL VALVE NUMBERS ARE PRECEEDED BY LP

FIGURE 3 LPCI SYSTEM DIAGRAM

REACTOR WATER CLEANUP SYSTEM

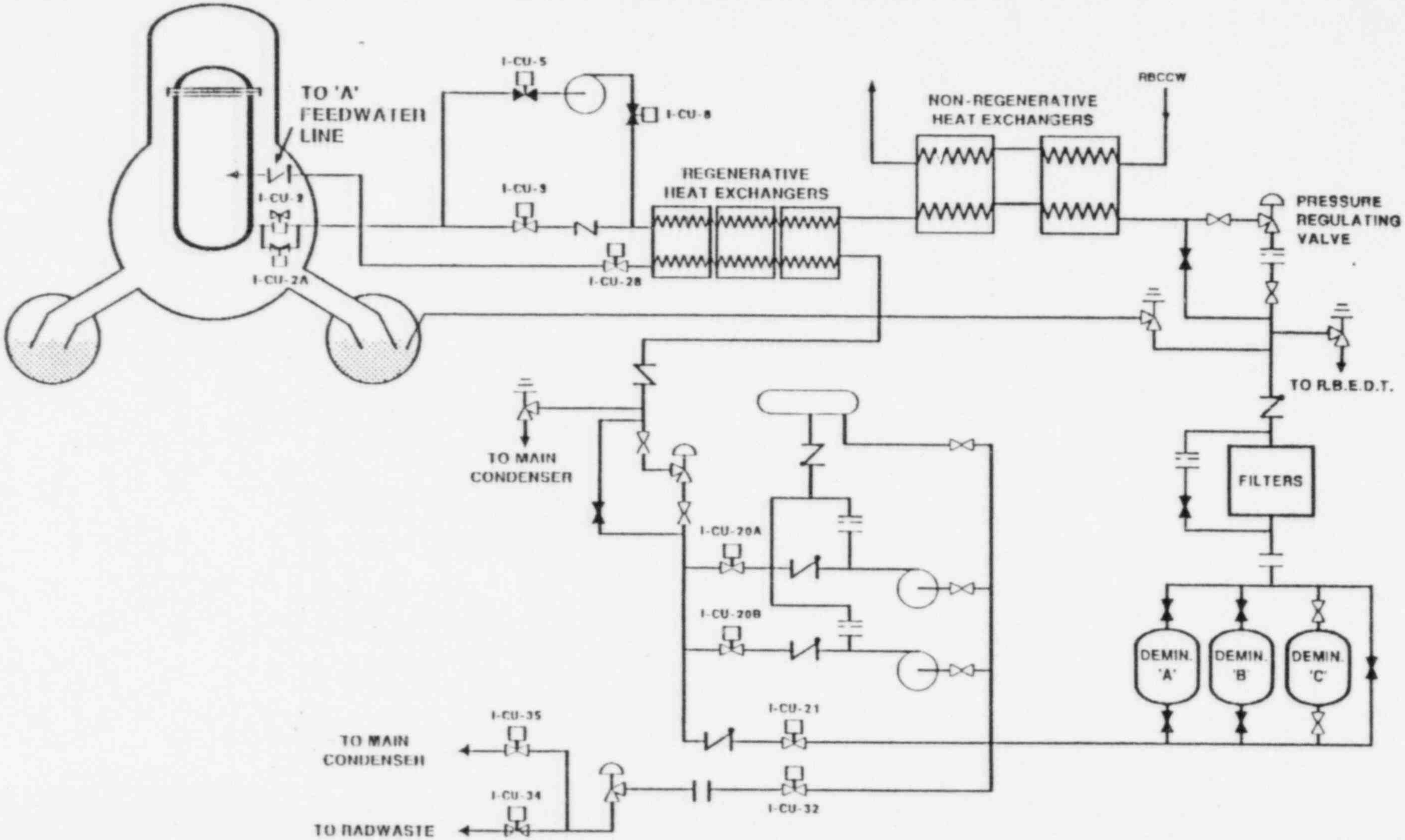


FIGURE 4