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December 6, 1996

Office of the Secretary
United States Nuclear Regulatory
Commission
Washington, D.C. 20055

ATT: Emile Julian, Esq.
Docketing & Service Branch

Dear Mr. Julian,

Enclosed for filing with the Commission and the new EDO please find two memoranda concerning the Vermont Yankee nuclear power station, docket number 50-271. My client, Citizens Awareness Network, Inc., requests that the Commission and EDO evaluate these documents, pursuant to 10 CFR 2.206, to see if enforcement action is warranted based upon the information contained therein.

Thank you for your prompt attention to this matter,

Sincerely,

Jonathan M Block

Jonathan M, Block
Attorney for Citizens Awareness Network, Inc.

enc./ Memoranda on Vermont Yankee Nuclear Power Station
with attachment

cc: Deborah B. Katz, President
CAN
P.O. Box 83
Shelburne Falls, MA 01370-0083

OFFICE OF THE SECRETARY
CORRESPONDENCE CONTROL TICKET

PAPER NUMBER: CRC-96-1232 LOGGING DATE: Dec 12 96

ACTION OFFICE: EDO

AUTHOR: JONATHAN BLOCK
AFFILIATION: VERMONT

ADDRESSEE: EMILE JULIAN, SECY/DSB

LETTER DATE: Dec 6 96 FILE CODE: IDR-5 VERMONT YANKEE

SUBJECT: FORWARDS FOR FILING, TWO DOCUMENTS CONCERNING THE VERMONT YANKEE POWER STATION---

ACTION: Appropriate

DISTRIBUTION: CHAIRMAN

SPECIAL HANDLING: 2.206 PETITION

CONSTITUENT:

NOTES: ENCLS TO: EDO...PROVIDED TO THE CHAIRMAN'S OFFICE FOR REVIEW PRIOR TO FORWARDING TO THE EDO

DATE DUE:

SIGNATURE: . DATE SIGNED:
AFFILIATION:

EDO -- G960950

M E M O R A N D U M

DATE: December 5, 1996

TO: Vermont State Nuclear Advisory Panel

FROM: Citizens Awareness Network

RE: CAN'S analysis of Vermont Yankee's July 23, 1996 presentation on RHR minimum flow valve

CC: U.S. NRC and general release

Summary of CAN's Analysis of VY's Presentation, "RHR Minimum Flow Valve Enforcement Conference," (July 23, 1996).[1]

VY spent much time in its presentation trying to demonstrate that the safety significance of the reported RHR problem was minimal, despite VY's repeated failures to identify and correct it. There may, however, be an outstanding issue of some significance. When VY changed the normal position of the valves to open, they may have introduced a unreviewed safety question. These valves probably perform a containment isolation function. According to the slides, the open valves will not close on loss of DC power and may not close via remote operation. It is not apparent that VY addressed the containment isolation function when they fixed their problem. In other words, repositioning the valves may only have switched problems rather than eliminating problems.

Vermont Yankee Appendix K LOCA Analysis

Problem:

The residual heat removal (RHR) system at Vermont Yankee consists of two loops. Each loop has two pumps that take suction from the suppression chamber. Each pump has a minimum flow line that connects to a common minimum flow header for each loop that returns flow to the

[1] Copy attached hereto. CAN hereby gratefully acknowledges the work of Nuclear Safety Engineer David Lochbaum, Union of Concerned Scientists, Washington, D.C., in preparing this memorandum.

suppression chamber. Each minimum flow header has a minimum flow valve (MOV-16A for loop A and MOV-16B for loop B).

In the low pressure coolant injection (LPCI) mode of RHR operation, all four pumps automatically start upon indication of an accident condition. When the pressure in the reactor vessel drops from around 1,000 psig during normal operation to around 400 psig during the accident, the LPCI injection valves automatically open to allow RHR to pump water from the suppression pool to the reactor vessel for core cooling. The safety analyses indicate that the reactor core will be adequately cooled if only one RHR loop is available.

In the event of a large break loss of coolant accident (LOCA), the reactor pressure decreases rapidly as reactor coolant spills out through the large broken pipe. If the LOCA is caused by the rupture of a small pipe, the reactor pressure will remain above 400 psig for a long time. The RHR pumps will have automatically started upon the accident signal, but the LPCI injection valves will remain closed. To protect the large RHR pumps, the minimum flow valves automatically open to allow flow to be re-circulated back to the suppression pool. When the LPCI injection valves open, the minimum flow valves automatically close to prevent flow from being diverted from cooling the reactor core.

In the original Vermont Yankee configuration, the RHR minimum flow valve (MOV-16A and MOV-16B) were normally closed. The Vermont Yankee staff identified that a single power supply failure would prevent both RHR minimum valves from operating. If this power supply failure occurred during a small break LOCA, all four RHR pumps would automatically start and run without a flow path for a long period of time until the LPCI injection valves opened. The concern was that all four RHR pumps might be damaged.

It is significant that Vermont Yankee personnel reported at least 10 "missed opportunities" to identify this problem over the 22 year lifetime of the plant.

Resolution:

The normal position of the RHR minimum flow valves was changed to open from closed. Since the valves still have the single failure vulnerability (i.e., now they will remain open if the single power supply failure occurs), Vermont Yankee personnel reanalyzed the small break LOCA event and determined that adequate core cooling would be provided even with some flow diverted to the suppression pool via the open minimum flow valve. This provides reasonable assurance that the facility can adequately handle a small break LOCA event in its current configuration.

To evaluate the consequences of the as-found configuration, Vermont Yankee personnel examined plant and industry experience and documented several instances where large pumps, including pumps identical to the Vermont Yankee RHR pumps, operated without minimum flow protection for periods of up to five hours without apparent damage. This provides reasonable assurance that the facility would have been able to handle a small break LOCA event in its original configuration even with the deficiency.

Potential Problem:

The RHR minimum flow valves (MOV-16A and MOV-16B) are shown on the slide "BACKGROUND SYSTEM DIAGRAM" to be the first isolation device on a line penetrating the suppression pool below the water line. Although it is not specified in the presentation slides, the RHR minimum flow valves probably have a containment isolation function. The RHR minimum flow valves at other boiling water reactor (BWR) plants of similar design (FitzPatrick, Browns Ferry, Peach Bottom, Hatch, etc.) perform a containment isolation function. Specifically, the RHR minimum flow valves must close to isolate the primary containment in the event of a pipe break outside containment.

When Vermont Yankee personnel changed the normal position of the RHR minimum flow valves from closed to open, they resolved the RHR pump protection problem. However, it is not certain that the containment isolation function of these valves was addressed. According to the slides "EXAMPLE CASE STUDY SINGLE FAILURE OF DC BUS 1 WITH LNP," the RHR minimum flow valves are normally open and fail open on loss of this power supply. With these valves now normally open, a single failure could prevent isolation of two containment penetrations.

In addition, the footnotes to these slides imply that the RHR minimum flow valves may not be provided with remote manual closure capability. Since the RHR minimum flow valves have dual function (to open on low pump flow for pump protection and to close for containment isolation), the operator in the control room must have the ability to override the automatic open signal to close the valves when necessary for containment isolation. In 1992, the New York Power Authority reported this exact deficiency to the NRC (LER 50-333/92-037-00 dated July 24, 1992).

In response to NUREG-0737 requirements, all licensees were required to evaluate containment isolation dependability following the Three Mile Island accident. It is imperative that the current configuration of the RHR minimum flow valves at Vermont Yankee be properly evaluated for containment isolation dependability, including remote manual closure capability.

Vermont Yankee Nuclear Power Corporation

RHR Minimum Flow Valve Enforcement Conference

July 23, 1996

Vermont Yankee
Nuclear Power Corporation

Attendees

Ross Barkhurst	President and CEO
Jay Thayer	Vice President Engineering
Bob Wanczyk	Plant Manager
Stan Miller	Design Engineering Manager
Jim Callaghan	Lead Fluid Systems Engineer
Michele Sironen	VY Nuclear Engineering Coordinator
Bruce Slifer	Senior Fluid Systems Engineer
Jim Duffy	Licensing Engineer

Enforcement Conference Agenda

• Introduction	Jay Thayer
• Background	Bruce Slifer
• Missed Opportunities	Stan Miller
• Corrective Action Process	
- Short-Term Corrective Actions	Jim Callaghan
- Root Cause Analysis	Bruce Slifer
- Single Failure Assessment	Jim Callaghan
• Safety Assessment	
- RHR Pump Performance	Stan Miller
- LOCA Analysis Impact	Michele Sironen
- IPE Impact	Michele Sironen
• Conclusion	Jay Thayer

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Introduction

Apparent Violation

Narrow focus: past process insufficient to detect/correct problems

Mitigating Factor

Current Corrective Action Process broad based and comprehensive

Introduction

Apparent Violation

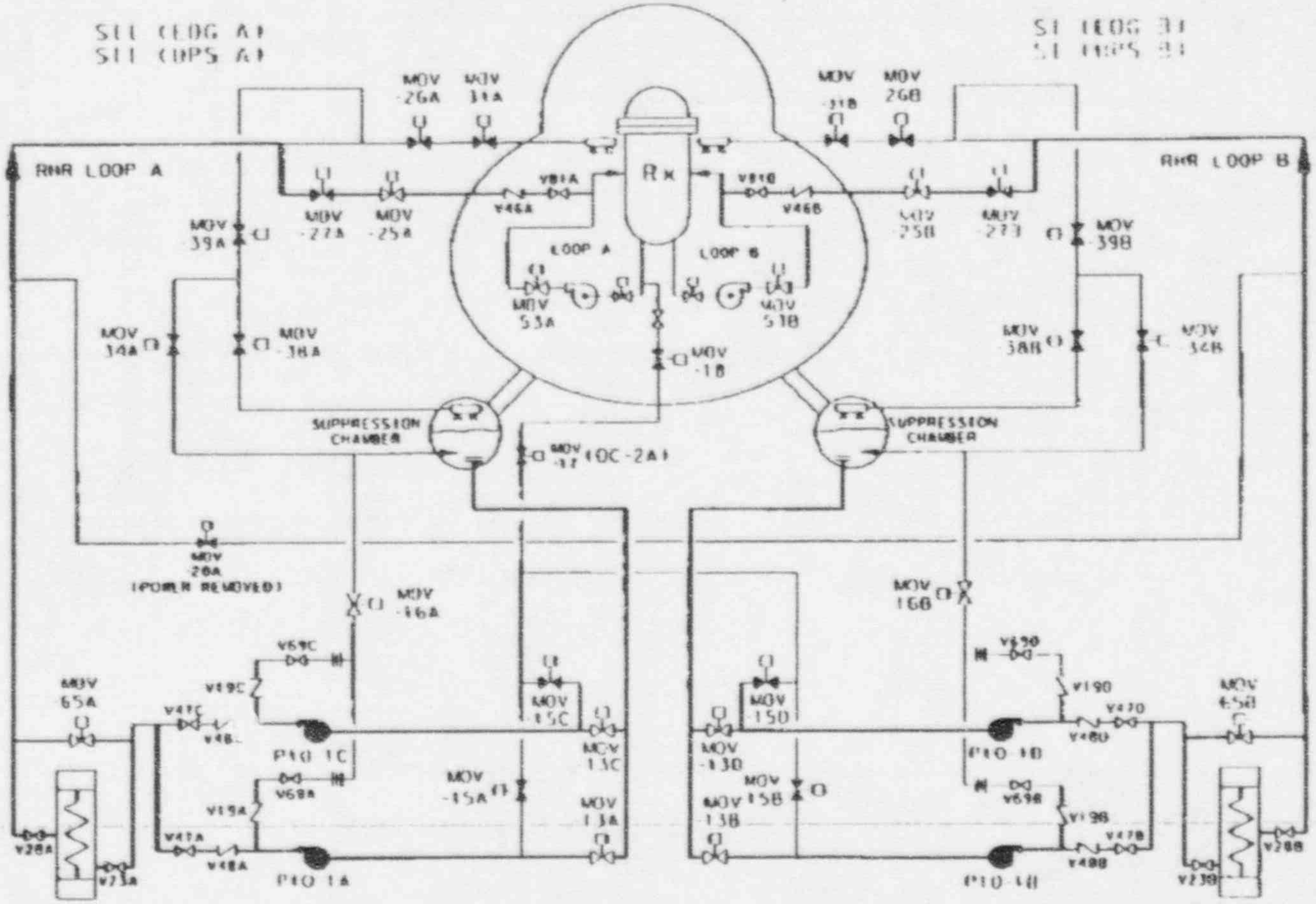
Failure to perform Appendix K, LOCA analysis assuming worst-case single failure

Mitigating Factors

- Self-identified
- Prompt and comprehensive corrective actions
- LOCA analysis had been performed with worst-case single failure known at the time
- Existing beyond design basis LOCA analysis bounded the identified worst-case single failure scenario
- This analysis shows that 10CFR50.46 limits would not be exceeded

Background

System Diagram



RHR A

RHR B

NOTE:

1. VALVE POSITIONS SHOWN FOR NORMAL LPCI MODE STANDBY ALIGNMENT

VERMONT YANKEE RESIDUAL HEAT REMOVAL SYSTEM
LOW PRESSURE COOLANT INJECTION MODE (LPCI)
CURRENT

Background

Appendix R Reanalysis Finding

- Event discovered as result of comprehensive review by fire protection team
- Review conducted as corrective action in response to Appendix R violation
- Cable-by-cable verification was in progress for all systems
- Cross-connects between Bus 3 and 4 identified
- Appendix R team identified single failure concern and promptly initiated event report process

Missed Opportunities

To Prevent:

- Failure to address vessel draindown concern (1971)
- Failure to change P&ID (1971)
- Failure to perform thorough single failure evaluation (1974)

Missed Opportunities

To Discover:

- NEDO-20967 single failure evaluation (1975)
- Generic evaluation of dc power failure (1978)
- IEB 86-01 minimum flow logic problems (1986)
- GE safety evaluation regarding minimum flow design adequacy (1987)
- LER 89-09 single failure results in loss of both RHRSW loops (1989)

Missed Opportunities

To Discover:

- **Plant Design Change Request (PDCR) 89-04**
- **Review of Cooper Significant Event Report (1993)**
- **New LOCA analysis with RELAP5YA (1993)**
- **Transfer IPE insights to design basis LOCA results (1993)**
- **P&ID corrective drawing update to show minimum flow valves closed (1995)**

Missed Opportunities

Fundamental Question

- **With the opportunities presented over 22 years, how did we fail to identify this mistake earlier?**

Missed Opportunities

- Need for questioning attitude in all activities not consistently reinforced
- Methods for performing event evaluations not always broad enough in focus
- Design basis documentation not easily retrieved

Program Improvements

- Improved questioning attitude/safety culture/external perspective
- Improve review process and set higher management expectations for evaluating operating events
- Significant improvements in engineering training programs
- Accelerate DBDs for high and medium safety significant systems consistent with IPE and Maintenance Rule
- Complete FSAR upgrade in 1997
- Developing System Engineering Program

Missed Opportunities

Summary

- Continued emphasis on questioning attitude and safety culture supported with external perspective

- Problem evaluation program has been improved over the years and is in the process of additional improvements

- Design basis documentation and FSAR upgrades to be accelerated

Corrective Action Process

Short-Term Corrective Actions

- Prompt event report and immediate operability assessment

- Basis for Maintaining Operation (BMO)

- Timely 10CFR50.59 safety evaluation for valve position change

Short-Term Corrective Actions

Event Report 96-229

- Initiated April 11, 1996 on LOCA single failure analysis concerns (1615)
- Initial design engineering assessment on operability concern
  - Pump vendor information limited break size concerns
  - Industry events of RHR pumps in no-flow condition
  - Analysis for one core spray pump and ADS (1993)
  - Discussed evaluation with plant management and NRC Resident on April 11, 1996

Short-Term Corrective Actions

Event Report 96-229

- Discussed concern with shift supervisor on April 11, 1996 (1710)
  - Shift supervisor determined, based on engineering evaluation, no immediate operability concern
  - Shift supervisor determined VY was outside design basis per 10CFR50.72(b)(1)(ii)B
  - Shift supervisor initiated non-emergency, one-hour notification (1724)

Short-Term Corrective Actions

Event Report 96-229

- Event report discussed at event report screening meeting on April 12, 1996
  - ER was determined to be a Level 1 event, the highest level available
  - Basis for maintaining operation was requested within a seven day time frame

Short-Term Corrective Actions

Event Report 96-229

- Event report presented to Plant Operations Review Committee (PORC) immediately following screening
  - Significance of finding led to special PORC presentation
  - Event discussed in length, with emphasis on operability concerns
  - PORC determined initial assessment acceptable, re-emphasized BMO time frame
- Formal root cause analysis initiated based on Level 1 event report

Short-Term Corrective Actions

Basis for Maintaining Operation (BMO 96-07)

- Initiated on April 12, 1996 to support operability assessment of event report
- Identified factors which compensated for adverse condition
  - Discussions with pump vendor of potential damage
  - Industry experience of pumps operating in a no-flow condition
  - Analysis for one core spray and ADS LOCA case
  - IPE analysis conclusion of low risk significance

Short-Term Corrective Actions

Basis for Maintaining Operation (BMO 96-07)

- Recommendations for correcting the condition
  - 10CFR50.59 initiated to support line-up change for minimum flow valves
- BMO 96-07 presented to PORC April 19, 1996
  - PORC concluded basis for operability was sound but additional documentation was needed
- BMO 96-07, Revision 1, presented to PORC April 25, 1996 and approved by Plant Manager on May 1, 1996

Short-Term Corrective Actions

10CFR50.59 Safety Evaluation

- Safety evaluation of changing normal position of minimum flow valve initiated
- Comprehensive design basis evaluation considered:
  - Original system design and minimum flow valve position
  - Impact of position change on current calculations (LOCA, long-term containment cooling)
  - Impact of position change on Vermont Yankee programs (Appendix R, Appendix J, ISI, IST, MOV)
  - Impact on plant controlled documents (FSAR, procedures)
  - Discussions with other utilities

Short-Term Corrective Actions

10CFR50.59 Safety Evaluation

- Safety evaluation was presented to PORC on April 24, 1996
  - Evaluation accepted with commitment to address changes required
  - Evaluation approved by Plant Manager on April 25, 1996
- Minimum flow valve normal position changed to open on April 26, 1996

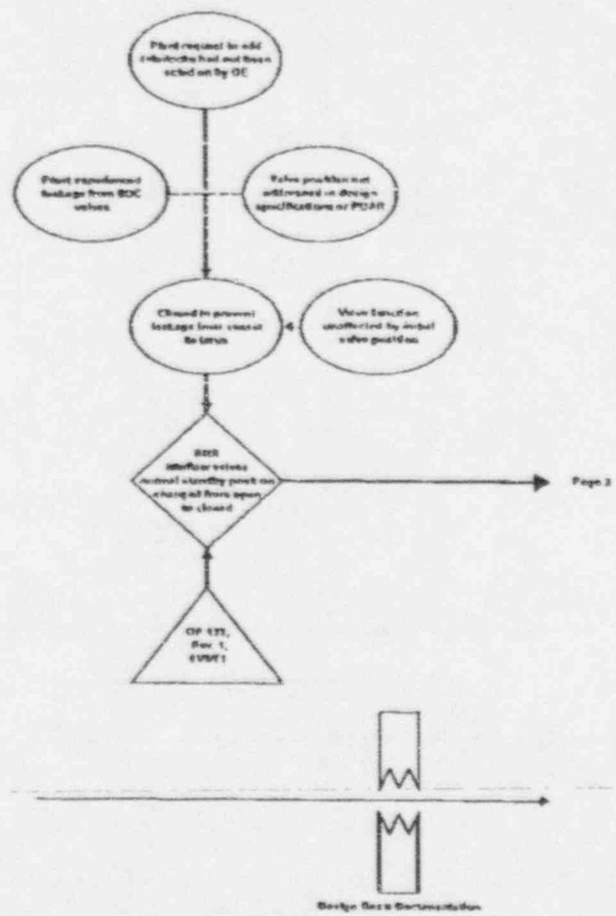
Short-Term Corrective Actions

Summary

- Prompt
- Immediate
- Comprehensive

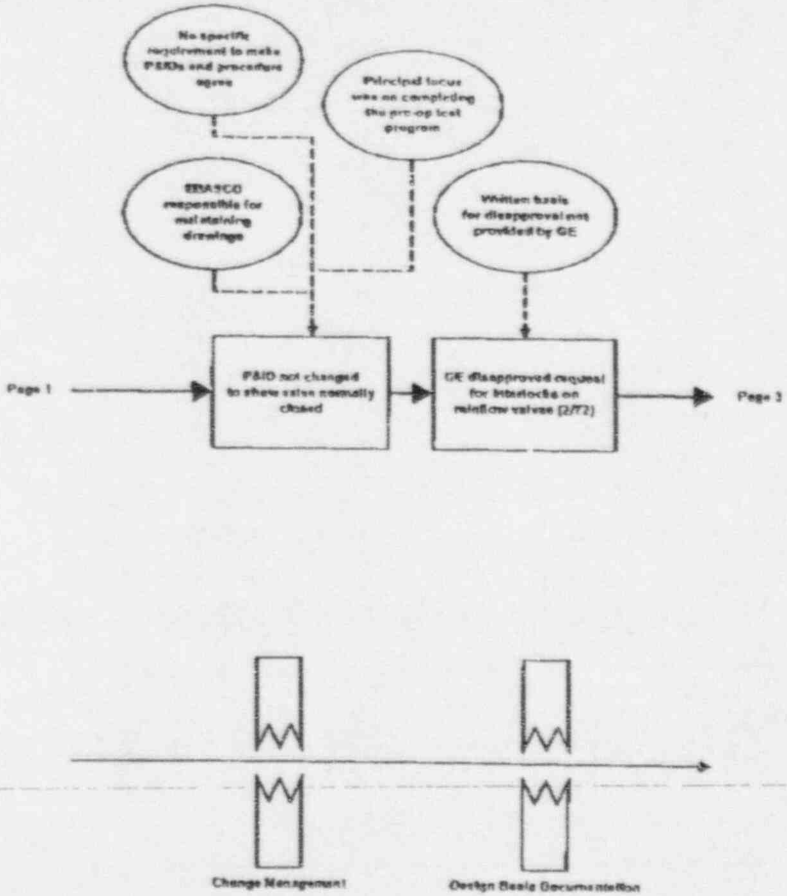
Root Cause Analysis

Loss of RHR Minflow Protection



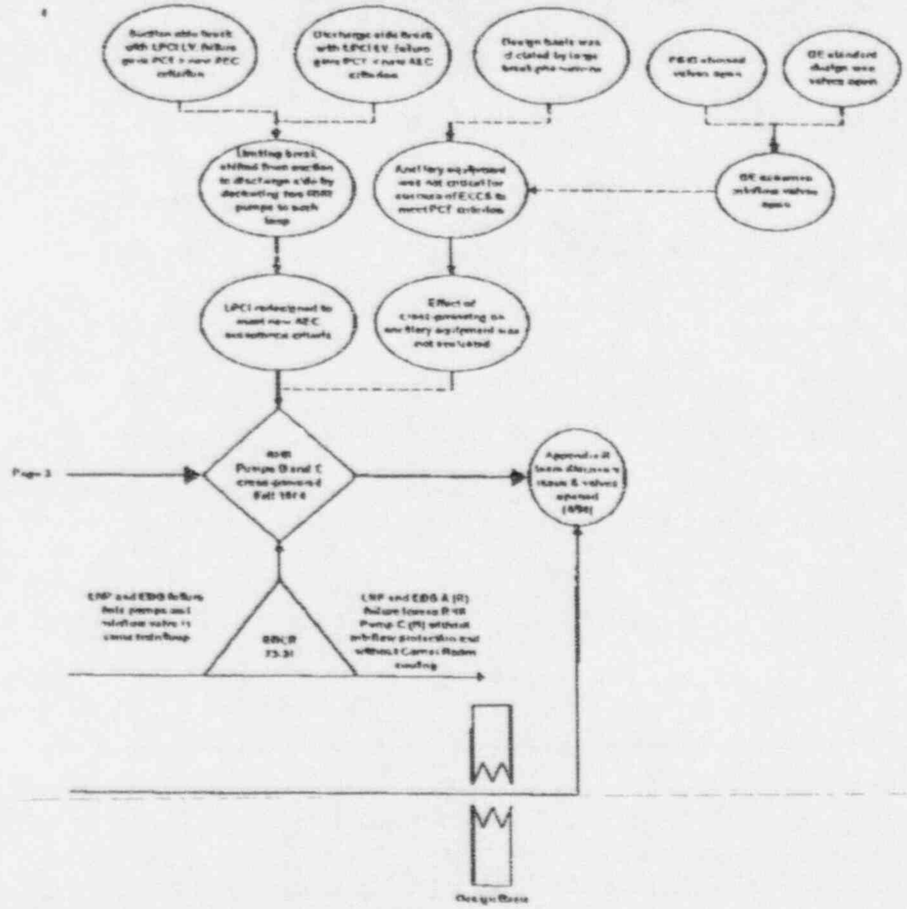
Root Cause Analysis

Loss of RHR Minflow Protection



Root Cause Analysis

Loss of RHR Minflow Protection



Root Cause Analysis

- Event Definition
  - Failure to provide minimum flow protection for cross-powered RHR pumps
- Primary Effects
  - Minimum flow valves closed by procedure, but P&ID unchanged
  - RHR Pumps B and C cross-powered but not minimum flow valves

Root Cause Analysis

- Root Cause
  - Inadequate design/single failure evaluation
- Contributing Causes
  - Inadequate documentation of minimum flow design basis
  - ECCS design focus on large break phenomena
  - Design engineers assumed minimum flow valves were open when they were closed

Root Cause Analysis

Corrective Actions - Completed

- Maintain minimum flow valves open during normal operation
- Revise LOCA analysis to account for failure of valve to close
- Review current design change process to ensure broad scope, comprehensive reviews

Root Cause Analysis

Corrective Actions - In Progress

- Complete single failure vulnerability review for all ECCS
- Revise FSAR description of minimum flow function
- Revise P&ID to show minimum flow valves open
- Develop policy on use of P&ID as design basis document

Root Cause Analysis

Corrective Actions - Planned

- Provide copies of RCA and missed opportunities to:
  - Task team working on OE assessment improvements (7/96)
  - Training Department for inclusion in ESP training (7/96)
  - Design engineering personnel (7/96)
- Self-assessment of process used to transfer analysis assumptions to operating procedures (12/96)
- Self-assessment of drawing revision process (12/96)

Root Cause Analysis

Summary

- Investigated event origins
- Identified root and contributing causes
- Corrective actions

ECCS Single Failure Assessment

- Purpose
  - Identify all single active failures
  - Confirm ECCS design basis LOCA requirements are satisfied

ECCS Single Failure Assessment

Scope

- Initial detailed assessment complete - undergoing independent review
- ECCS systems/components
  - HPCI
  - ADS
  - CS
  - LPCI
- Electrical systems
  - 125V/24 Vdc power
  - 4 kV/480 Vac power
  - 480 V/LPCI-UPS power

ECCS Single Failure Assessment

Scope

- Short-term ECCS injection (blowdown, refill, reflood)
- No credit for ECCS delivery to broken loop/pipe
- Coincident loss of off-site power

ECCS Single Failure Assessment

Bounding Single Active Failure Cases

- DC-1 bus failure
- DC-2 bus failure
- LPCI UPS failure

Example Case Study
Single Failure of DC Bus 1 with LNP

SINGLE ACTIVE FAILURE AVAILABLE ECCS SYSTEMS	No Break	Break Location: Recirc. Discharge Line, Loop A	Recirc. Discharge Line, Loop B	Recirc. Suction Line, Loop A	Recirc. Suction Line, Loop B	Core Spray Line	Feed Water Line	Main Steam Line
DC-1 Failure w/ Loss of Normal Power (LNP)	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes
INDEPENDENT SYSTEMS AVAILABLE FOR SHORT TERM ECCS INJECTION MODE (with DC-1 failure)								
Division B2 DC Power via Battery B-1	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes
Division S2 AC Power via EDG-1A/Bus 4	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes
ECCS Logic Division A	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes
RCIC via DC-2	Yes	Yes	Yes	Yes	Yes	Yes	No	No
ADS: - Logic via Division A - 4 SRVs powered from DC-2C	 Yes Yes	 Yes Yes	 Yes Yes	 Yes Yes	 Yes Yes	 Yes Yes	 Yes Yes	 Yes Yes
ADS Availability:	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes
LPCI Loop A: - Pump P-10-1A via EDG-1A/Bus 4 * MOV-16A Min. flow valve (N.O., A.C.) * RRU-7 NE Corner Room Cooler - Loop A Valves w/Power from MCC-89A (LPCI MG-Set-1A) and ECCS Div. A Logic * Inj. Vlv MOV-25A (N.O., A.O.) * Inj. Vlv MOV-27A (N.C., A.O.) * Recirc. Discharge MOV-53A (N.O., A.C.) * Recirc. Bypass MOV-54A (N.O., A.C.)	 Yes Yes Yes Yes Yes Yes Yes	 Yes Yes Yes Yes Yes No No	 Yes Yes Yes Yes Yes Yes Yes	 Yes Yes Yes Yes Yes No No	 Yes Yes Yes Yes Yes Yes Yes	 Yes Yes Yes Yes Yes Yes Yes	 Yes Yes Yes Yes Yes Yes Yes	 Yes Yes Yes Yes Yes Yes Yes
LPCI Loop A Availability:	Yes	No	Yes	No	Yes	Yes	Yes	Yes
LPCI Loop B: - Pump P-10-1B via EDG-1A/Bus 4 * MOV-16B Min. flow valve (N.O., F.O.) * RRU-8, SE Corner Room Cooler - Loop B Valves w/power from MCC 89B (LPCI MG-Set-1B) and ECCS Div. A Logic * Inj. Valve MOV-25B (N.O., A.O.) * Inj. Valve MOV-27B (N.C., A.O.) * Recirc. Discharge MOV-53B (N.O., A.C.) * Recirc. Bypass MOV-54B (N.O., A.C.)	 SI Yes No Yes Yes Yes Yes	 SI Yes No Yes Yes Yes Yes	 SI Yes No Yes Yes No No	 SI Yes No Yes Yes Yes Yes	 SI Yes No Yes Yes No No	 SI Yes No Yes Yes Yes Yes	 SI Yes No Yes Yes Yes Yes	 SI Yes No Yes Yes Yes Yes
LPCI Loop B Availability:	SI	SI	No	SI	No	SI	SI	SI

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Example Case Study
Single Failure of DC Bus 1 with LNP

SINGLE ACTIVE FAILURE AVAILABLE ECCS SYSTEMS	No Break	Break Location: Recirc. Discharge Line, Loop A	Break Location: Recirc. Discharge Line, Loop B	Break Location: Recirc. Suction Line, Loop A	Break Location: Recirc. Suction Line, Loop B	Break Location: Core Spray Line	Break Location: Feed Water Line	Break Location: Main Steam Line
CS Train A:								
- Pump P-46-1A (via E06-1A/Bus 4)	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes
* MOV-11A (N.O., R.O.)	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes
* MOV-12A (N.C., A.O.)	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes
* MOV-5A Min. flow valve (N.O., A.C.)	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes
* RRU-7, NE Corner Room Cooler	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes
CS Train A Availability:	Yes	Yes	Yes	Yes	Yes	No(1)	Yes	Yes
INDEPENDENT SYSTEMS AVAILABLE FOR LONG TERM(13) TORUS COOLING MODE (with DC-1 failure)								
RHR/Torus Cooling Loop A via EOPs:								
- RHR/LPCI Pump P-10-1A	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes
* MOV-16A Min. flow valve (N.O., A.C.)	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes
* RRU-7, NE Corner Room Cooler	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes
* MOV-65A (N.O., R.M.C.)	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes
* MOV-34A (N.C., R.M.O.)	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes
- RHRSW Pumps P-8-1A and P-8-1C	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes
* MOV-89A (N.C., R.M.O.)	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes
* RRU-5, NE Corner Room Cooler	Yes(10)	Yes(10)	Yes(10)	Yes(10)	Yes(10)	Yes(10)	Yes(10)	Yes(10)
- SW Pumps P-7-1A and P-7-1C, and SW MOV-20	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes
RHR/Torus Cooling Loop A Long Term Availability:	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes

Notes:
(1) LOCA analysis assumes no credit for ECCS delivery to broken loop.
(2) LOCA analysis assumes no credit for closure of recirc. loop discharge valve in broken loop for discharge and suction line breaks.
(3) HPCI available, but not credited in LOCA analysis.
(4) HPCI assumed failed due to break location in steam line. Not credited in LOCA analysis.
(5) RCIC available, but not credited in LOCA analysis.
(6) RCIC assumed failed due to break location in steam line. Not credited in LOCA analysis.
(7) Performance of SRV on broken MSL may be degraded due to break location.
(8) Available for short term (ST) LPCI injection mode only. Injection from this path credited in LOCA analysis.
(9) Min. flow valve fails open, no auto closure on high flow. Analysis assumes valve remains open and causes small flow diversion.
(10) Room cooler RRU-5 not needed when RRU-7 is operating.
(11) Room cooler RRU-6 not needed when RRU-8 is operating.
(12) Short term (ST) refers to post accident period including vessel blowdown, lower plenum refill, and core reflood.
(13) Long Term (LT) refers to post accident period including sustained core cooling w/ clad temperature near Tsat.

N.C./R.C. = Normally Closed/Remain Closed
N.O./R.O. = Normally Open/Remain Open
A.O./A.C. = Auto-Open/Auto-Close
F.O./F.C. = Fail Open/Fail Closed
R.M.O./R.M.C = Remote Manual Open/Remote Manual Close

PNRLETYJIC/NTC-NFTG3)

ECCS Single Failure Assessment

Summary

- Did not identify any new single active failures
- ECCS availability matched design basis LOCA analysis

Safety Assessment

RHR Pump Performance

- Immediate operability assessment
- A PWR operated an RHR pump 66 minutes in a no-flow condition without damage
- Another PWR tested a spare RHR pump in a no-flow condition for approximately one hour without damage
- A BWR operated an RHR pump in a no-flow condition for more than one hour without damage

Safety Assessment

RHR Pump Performance

- Sister utility ran RHR pump for five hours in no-flow condition (1981)
- Pump remained operable
- Pump tested three days later - no degradation in head/capacity
- Pump test showed low vibration levels giving assurance of no mechanical damage
- Pump was of same manufacturer and model as Vermont Yankee pumps

Safety Assessment

LOCA Analysis Impact

- Original ECCS design basis analysis relied only on CS and ADS
- SAFE/REFLOOD ECCS analysis results (intermediate-to-large breaks) unaffected by availability of RHR pump
- Realistic ECCS analyses using SAFE for small breaks with one CS System + two ADS valves: PCT < 2200°F
- RELAP5YA-BWR beyond design basis analysis - one core spray + ADS: maximum PCT = 1806°F
- RELAP5YA-BWR design basis analysis - current, RHR minflow valve open: maximum PCT = 1793°F
- With current analysis methods (RELAP5YA-BWR) 10CFR50.46 limits would not be exceeded due to loss of all LPCI flow

Safety Assessment

IPE Impact

- IPE correctly modeled RHR minimum flow valves closed
- IPE assumed closed valves would fail RHR pumps for small and medium LOCA events
- Overall CDF of 4E-6 not affected by discovery of design basis issue
- All LOCAs are 2% of CDF
- RHR pump failures due to minimum flow valves did not appear in any LOCA sequence >1E-9

Safety Assessment

Summary

- RHR pumps are expected to work
- ECCS criterion is met
- IPE CDF is not impacted

Enforcement Considerations

Identification

- Self-identified through comprehensive corrective actions from a previous enforcement action (Appendix R)
- Not identified through an event
- Involves an old design issue

Enforcement Considerations

Corrective Actions

- Prompt and comprehensive corrective actions
  - Immediate operability assessment
  - Basis for maintaining operation
  - 10CFR50.59 evaluation
  - Licensee event report
  - Comprehensive root cause analysis
  - Extensive management oversight

Conclusion

Design Basis/Engineering Program Review

- Threshold for findings low
- Actions on findings more consistent
- Management reinforcement of questioning attitude

M E M O R A N D U M

DATE: December 6, 1996

TO: Vermont State Nuclear Advisory Panel

FROM: Citizens Awareness Network, Inc.

RE: Vermont Yankee Nuclear Power Station 1996 Licensee Event Report Summary (May 1996 - October 1996).

CC: U.S. NRC and general release

Citizens Awareness Network's Critical Analysis[1] of Vermont Yankee Nuclear Power Station's 1996 Licensee Event Report Summary (May 1996 - October 1996)

LER 96-13

Vermont Yankee's analysis:

"The LER describes a fire protection issue that was identified during our Fire Protection Program self-assessment. We identified a section of piping that was not included in the design calculation for the fire sprinkler system. The concern was that the water flow to the sprinklers could be less than expected if all the sprinklers activated simultaneously. Although such as activation is not likely, we are recalculating the sprinkler system design for such a scenario."

CAN's analysis:

This LER describes TWO fire protection issues: (1) the reactor recirculation pump motor generator foam suppression system, and (2) the reactor building sprinkler system at elevation 252'. Neither the LER nor Vermont Yankee's analysis address the cause and consequences of the foam suppression system deficiency.

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[1] CAN gratefully acknowledges the assistance of David Lochbaum of Union of Concerned Scientists, Washington, D.C., in preparing these critical analyses.

The LER attributes the root cause to personnel errors by the contractor who designed and installed the sprinkler system. LER 96-20, which documents another fire protection deficiency, indicates that the Vermont Yankee Cognizant Engineer failed to adequately review the design package which the contractor developed in that case. Vermont Yankee also failed to adequately review the sprinkler system design package that is the subject of LER 96-13.

LER 96-14

Vermont Yankee's analysis:
"As a result of a concern identified at another plant, we initiated a review of the atmospheric venting capability of our diesel generator rooms. The concern relates to the possible affect on the operability of the diesels due to significant changes in atmospheric conditions inside the rooms resulting from a tornado. Our assessment determined that we needed to improve the venting capability of the rooms."

CAN's analysis:
The LER indicates that Vermont Yankee's FSAR has been incorrect with respect to the design features necessary to protect the diesel generator rooms during a tornado. Although the LER indicates that Vermont Yankee may have intended to install appropriate protective devices, it never did so. This long standing deficiency raises serious questions about the safety analyses--such as the Individual Plant Examination--which VY performs to support continued operation of the nuclear power station. The FSAR is used in extremely significant ways relating to safe operation. For example, the FSAR is used as an input document for safety evaluations performed under 10 CFR § 50.59, in the development and validation of the plant simulator, in the development of emergency response procedures, and many other vital activities. Hence,

flaws in the FSAR cause serious, rippling effects throughout VY's safety systems. In cases like those presented by this LER, Vermont Yankee must include assessments of the impact of the deficient conditions upon all affected programs. Only in this way can VY accurately determine the general effect of such deficiencies upon safe operation of the Vermont Yankee nuclear power station.

Vermont Yankee states that there have been no previous LERs reported to the NRC regarding failure to satisfy design basis tornado protection in the past five years. If this statement refers to LERs submitted by all licensees and not just those submitted by Vermont Yankee, then it is in error. The New York Power Authority submitted an LER for a similar problem at their James A. FitzPatrick Nuclear Power Plant in 1992/1993.

LER 96-15

Vermont Yankee's analysis:
"In response to an issue identified at another plant, we performed an engineering evaluation of certain piping systems. The evaluation looked at the potential for piping to become overpressurized under certain accident scenarios due to thermal induced pressurization. Pressure relief ports were installed on some sections of piping during the refueling outage to address this concern."

CAN's analysis:
The LER deals with inadequate thermal protection for piping lines in six systems dating back to original construction of the facility. VY attributes the root cause to personnel errors during plant design. Thus, VY fails to point out that every modification to these six systems since construction represents a missed opportunity to identify and correct the original errors.

LER 96-18

Vermont Yankee's analysis:
"As a result of the Fire Protection Program self-assessment, we identified a concern with the adequacy of the fire wrap for a small section of electrical cable. The concern was that the configuration of this particular section of fire wrap was different than the manufacturers tested configuration. The fire wrap was modified so that it meets the tested configuration."

CAN's analysis:
The LER documents a deficiency with significant adverse safety implications. A fire could have rendered inoperable both divisions of the emergency core cooling systems. According to the LER, this problem existed for nearly a decade!

LER 96-19

Vermont Yankee's analysis:
"This LER addresses the inadvertent trip of a breaker in our reactor protection system. Our review found that the cause was a loose electrical connection. The breaker was repaired and returned to service."

CAN's analysis:
The LER involves a random equipment failure that was corrected As stated in the LER, this event presented no significant risk to public health and safety.

LER 96-20

Vermont Yankee's analysis:
"This LER resulted form the self-identification of a concern with the adequacy of how we tested the CO^2 fire suppression system for our Switchgear Room. We found that we should have adjusted the testing method following the implementation of design modifications we made to the room."

CAN's analysis:
The LER involves a problem resulting from VY's modification to the plant in 1978! As VY states in

the LER and discusses in its analysis, VY's subsequent testing failed to identify the design fault. In 1982, VY implemented a modification which affected the CO^2 system without conducting any post-modification testing. In this way, <u>the LER reveals a serious deficiency in VY's design change control process</u>. Prudent practice dictates that VY should perform an "extent of condition" evaluation to determine how many other modifications have been inadequately tested since startup.

LER 96-21

<u>Vermont Yankee's analysis</u>:
"The LER discusses the unexpected trip on one of our residual heat removal pumps. The cause of the trip was related to the pump's control circuitry. The circuitry was adjusted and the pump was returned to service."

<u>CAN's analysis</u>:
The LER indicates that Vermont Yankee personnel conducted a thorough investigation to determine if other equipment had the same problem that caused the RHR pump to trip. As VY states in the LER, there was no significant risk to public health and safety.

LER 96-22

<u>Vermont Yankee's analysis</u>:
"This LER describes the self-identification of an inoperable breaker associated with one of the diesel generators. During routine walkarounds, an operator noticed a couple of small screws on the floor. He immediately checked equipment in the area and determined that the breaker was inoperable. The breaker was repaired and all other similar breakers were inspected."

<u>CAN's analysis</u>:
The LER states:

> As the B EDG [emergency diesel generator] and the emergency Alternating Current (AC) power source were at all times available during this event, and the failure of a single EDG is bounded by our current [nuclear power] plant accident and transient analyses, this event did not result in plant operation which endangered the health or safety of the public.

This statement is ill-founded.

While it may be true that the Vermont Yankee nuclear power station accident and transient analyses demonstrate reasonable assurance that public health and safety will be protected in event of failure of a single EDG, it is improper for VY to rely on this argument to justify an EDG failure lasting longer than the Technical Specification duration. <u>Vermont Yankee's Technical Specifications limit that duration to only 7 days</u>. The <u>VY nuclear reactor continued to operate, while the failed EDG went undetected for 25 days</u>! If an accident or transient had occurred when the EDG failed, public health and safety would have been reasonably assured ONLY if no other equipment failure occurred. Vermont Yankee tries to justify this flagrant breach of safe practice by stating that the remaining EDG and the emergency AC power source were operable. In order to maintain safe operation during accident conditions, ALL of the other emergency equipment (pumps, motors, valves, etc.) must respond flawlessly. Hence, it is specious of VY to attempt to justify continued operation without both EDGs fully operable.

Vermont Yankee once again misconstrues the function of Technical Specification limits upon the duration for which equipment may be out of service. The 7 day limit for the EDGs is intended to provide reasonable assurance that ALL equipment will be operable and available to respond in the event of an emergency. Vermont Yankee has improperly attempted to explain away the seriousness of this deficiency on the grounds that the analyses assume only a single failure. As a point of fact and law, the analyses assume that a single failure will occur during VY's response to the event, not that a pre-existing failure will start the accident.[2]

Once the problem was identified, Vermont Yankee personnel conducted a thorough "extent of condition" evaluation that identified and corrected similar problems affecting other breakers. However, VY's misconstruing the purpose of TS limits on equipment outages appears to be part of a chronic pattern of misunderstanding the underlying safety purposes of Technical Specifications, the FSAR, design bases, and NRC regulations.

LER 96-23

Vermont Yankee's analysis:
"During a routine surveillance of a radiation monitor, a technician determined that an output contact of the monitor had not been properly tested. Following a root cause analysis of the finding, we modified the related surveillance and test procedure to more clearly describe the testing method."

CAN's analysis:
As stated in the LER, the radiation monitor was demonstrated to be operable when re-tested using the corrected procedure. Therefore, this event

[2] *See* LER 96-25 for a related misapplication.

presented no significant risk to public health and safety.

LER 96-25

Vermont Yankee's analysis:

"The LER describes the self-identification of a concern with the testing criteria we used for a valve in the plant's nitrogen purge system. The engineering review found that the leak testing method we used for this particular valve needed to be adjusted. The valve was tested using the new methodology during the recent refueling outage."

CAN's analysis:

This LER reports a substantial deficiency in a containment isolation valve at the Vermont Yankee nuclear power station. The LER states that, "[l]eakage was beyond the capacity of the test device's capability to measure." This deficient condition existed since at least 1978! The LER concludes that this event presented no significant risk to public health and safety because a review "...show[ed] that since 1978 the applicable pathway leak rates were within TS limitations and accident assumption values....demonstrat[ing] that the containment system was at all times intact, providing a viable fission product barrier, consistent with plant design."

Once again, Vermont Yankee's conclusion is absolutely wrong and demonstrates a complete lack of appreciation for the "defense in depth" philosophy underlying NRC regulations.

The deficient containment isolation valve is the first of two containment isolation valves on TWO separate primary containment penetrations. If either of the two downstream containment isolation valves[3] failed to close during an accident, the excessive leakage through the deficient valve would not be contained. Such a failure has potentially serious, adverse public health consequences. The plant design requires that primary containment integrity be assured in the event of a single failure. Primary containment integrity at the Vermont Yankee nuclear reactor could not be established under such conditions.

Vermont Yankee yet again fails[4] to properly apply the "single failure" criterion when evaluating the significance of incidents at the nuclear power station. VY must not be permitted to accept pre-existing deficiencies--particularly those of a protracted duration--in its nuclear safety analyses. The analyses assume a single failure concurrent with the accident or subsequent to the accident, not single failures existing prior to the accident.

CONCLUSIONS

Based upon the foregoing analyses, CAN concludes that:

1. The number of long standing deficiencies VY acknowledges in these LERs--10 to 18 years!!--raise serious questions about the adequacy of VY's safety analyses. VY and the NRC should immediately begin reviewing all of the safety analyses conducted since startup of VY, with particular attention to their role in providing a complete and up-to-date FSAR.

2. VY needs to correct serious deficiencies in its design change control process. VY should immediately commence a review of its design

[3] Valves 23 and 9 on the figure provided in the LER.

[4] *See* LER 96-22 for the other instance.

control process, including an historical review of its design control documentation to verify the accuracy of this documentation when compared with the actual, physical configuration of VY.

3. VY should perform a global "extent of condition" evaluation to determine how many modifications have been inadequately tested since startup. As a corollary, any and all untested (or long ago tested) systems at VY should be immediately tested (or re-tested).

4. VY needs to initiate an thorough retraining program to review and emphasize the underlying <u>safety</u> purposes of Technical Specifications, the FSAR, design bases, and NRC regulations in relation to routine operation of the nuclear power station, emergency preparedness, and practical implementation of the NRC's "defense in depth" philosophy.

5. CAN strongly recommends that the Vermont Yankee staff receive training on the proper use of the "single failure" criterion.