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

July 9, 1996

MEMORANDUM TO: All Field Inspectors

FROM: James L. Milhoan James L. Milhoan
Deputy Executive Director
for Nuclear Reactor Regulation,
Regional Operations and Research

SUBJECT: ARTICLE IN CHRISTIAN SCIENCE MONITOR

I would like to direct your attention to the attached article which appeared in the Christian Science Monitor on June 18, 1996. The concerns raised in this article relate to the effectiveness of the NRC and they point to areas which we may need to improve. To this end, I would sincerely appreciate your perspective on the issues raised in the article, as well as your comments and views regarding your personal experiences, not only in communicating with your management, but also in your ability to perform your job.

Although your response is not mandatory, the issues raised in this article are relevant to each inspector in the field; thus, your input would be useful in identifying areas in need of improvement. Please provide your response to your respective Regional Administrator with a copy to me and to the Director of Inspection and Support Programs/NRR by August 30, 1996. It is my intent to have the Regional Administrators review your responses and discuss them at the January 1997 Senior Management Meeting in Region IV.

Attachment:
As stated

The Christian Science Monitor
United States
Tuesday June 18, 1996 Edition

Walking the Beat With a Nuclear Patrolman

Peter N. Spotts, Staff writer of The Christian Science Monitor

SEABROOK, N.H. -- John Macdonald strides into the control room of Seabrook Station, a nuclear plant on the coast north of the Massachusetts-New Hampshire border.

He pauses to look at the dials and displays in this cockpit for controlling one of the world's most dangerous technologies.

"If you want to know what's really going on at a nuclear plant, just ask a reactor operator. They'll talk your ear off," he says, grinning, as one approaches.

CONTROL ROOM CONFERENCE: Senior resident inspector John Macdonald (r.) performs a routine check on operations at the Seabrook nuclear plant in New Hampshire. (PHOTOS BY R. NORMAN MATHENY/STAFF)

Mr. Macdonald is keenly interested in what the operators have to say - as well as all the digits on the monitors here. He is the Nuclear Regulatory Commission's senior resident inspector at the plant. As such, he is one of 181 on-site inspectors at 110 commercial plants across the country.

They are the agency's cops on the beat, the first line of defense against nuclear catastrophe. They prowl corridors and peer from catwalks, listening, watching, and asking probing questions.

How well they do their job - and the conflicts they face - go to the heart of the debate over the effectiveness of the NRC itself.

Certainly being a resident inspector is one of the more unusual jobs in government. Unlike many other federal watchdogs, resident nuclear inspectors go to work every day with the people they are supposed to oversee.

They have offices at the plant. They eat in the company cafeteria. Though federal rules forbid them from "socializing" with plant workers, they have to develop a level of trust with utility managers and staff while maintaining a sense of detachment.

Tensions can surface even with their own NRC superiors. Some on-site inspectors say they're hampered with by-the-book administrative work that eats into time better spent inspecting pumps and pipes. Other inspectors complain of supervisors altering or ignoring their findings. They cite instances of being harassed for pursuing safety issues by a senior management too cozy with the nuclear industry.

The result, critics say, is an agency in which dissent is often stifled and a nation in which reactors may be operating with defective systems.

While resident inspectors lack the authority to slap an errant power plant with a fine or even a notice of violation, the NRC's equivalent of a ticket, they are responsible for providing an independent check on plant performance.

Their reports cover everything from the nuts-and-bolts of plant repairs to reviewing documents to see how well operators identify and solve equipment problems. Inspectors keep tabs on how plants respond to NRC safety directives. They also serve as a representative to the public living near a nuclear facility - for instance, giving talks in local schools.

For his part, Macdonald says his experience as an inspector has been a good one. To spend time with him is to glimpse the magnitude of the job the NRC faces in regulating a technology in which there is little room for error.

"You've got something the size of Shea Stadium you've got to inspect," says one official at NRC headquarters. "You can't be on top of everything. You hope you're dealing with a responsible licensee."

By 8:30 on this morning, Macdonald, dressed casually in khaki pants and knit shirt, has already checked control-room operating records and taken part in a conference call with the NRC's regional headquarters in King of Prussia, Pa.

ON PATROL: Seabrook inspector John Macdonald is responsible for seeing that the utility safely operates a maze of pumps and generators, as well as miles of pipes and electrical cables that make up an atom-splitting reactor.

Moments later, he slips into a corner seat in a conference room as some 30 Seabrook officials and staff gather for a daily briefing. One by one, they review the plant's performance in the past 24 hours and report the status of maintenance projects. Macdonald jots notes as one describes a problem he's found with a radiation monitor. The malfunction doesn't seem to be serious, but the utility will need NRC approval to fix it.

Later, in his office, Macdonald says that the utility was in effect proposing the NRC adopt new restrictions on the way the plant operates. "That represents a good safety ethic," he adds.

The NRC, in fact, has to rely a lot on the integrity of utilities. That's because the agency "does not have the

manpower to come in and see that every 't' is crossed and every 'i' is dotted," says William Jocher, a former executive of the Tennessee Valley Authority who raised safety issues about the utility's Sequoyah nuclear plant.

Indeed, the amount of work involved raises an enduring question: Does the NRC have enough nuclear beat cops? At present, the agency matches the number of reactors at a site with the same number of resident inspectors, then adds one more. The so-called n+1 formula represents a compromise. In 1981, two years after the accident at Three Mile Island in Pennsylvania, the agency wanted to increase the number of resident inspectors from at least one per reactor to two. Congress balked at the cost.

Inspectors must have science or engineering backgrounds, although not necessarily in the nuclear field. Many come from the Navy or from nuclear utilities. Macdonald, one of two inspectors at Seabrook, came to the agency after getting a degree in marine engineering and a US Coast Guard license for operating steam, diesel, or gas turbines.

He joined the NRC in 1984 and arrived at Seabrook in April 1995, after serving as senior resident inspector at the Pilgrim nuclear plant in Plymouth, Mass., and as a resident inspector at plants in Vermont and Florida. His junior colleague, David Mannai, came to Seabrook from the Susquehanna nuclear plant in Pennsylvania. On this day, Mr. Mannai is away taking part in a six-week training program in Tennessee to get the basics on Seabrook's reactor, which is different from Susquehanna's.

Resident inspectors serve at one plant for up to five years. The rotation policy is designed to ensure that the watchdogs don't get too close to the people they're overseeing. Yet because nuclear facilities are complex assemblages of pumps, pipes, turbines, and emergency generators, it can take an inspector several years just to learn everything about a plant. "On average it takes one to two years to really get a handle on a site with any confidence," Macdonald says.

Ordinarily, Macdonald says, he and Mannai meet early in the morning in their office at the plant to set inspection agendas. Inspectors have a core set of plant activities they must scrutinize regularly. These range from plant maintenance and engineering to radiation exposure and security. The core program also includes "initiative" inspections - ones where "you have an itch, a hunch, or some aspect of the plant hasn't been reviewed in awhile," Macdonald says.

> **'If you polled every senior resident inspector out there, they'll tell you the same thing: [NRC] management won't let us do our job.'**
> ***-A 20-year NRC inspector***

In following this routine, inspectors will do everything from watch workers make repairs to review plant records to see how quickly a utility spots and fixes a problem. NRC guidelines are detailed - down to recommending how much time inspectors can devote to each inspection category. The idea is to catch problems before they become too big.

YET such micromanaging can also serve as a straitjacket, some inspectors say. "If you polled every senior resident inspector out there, they'll tell you the same thing: [NRC] management won't let us do our job," says a 20-year NRC veteran who is a resident inspector in another region.

Managers at his regional headquarters, he explains, keep close tabs on the inspection numbers in part because inspection time is factored into the fees the NRC charges utilities for its work. These fees pay the NRC's overall bill.

"I've had to use every goofy little process for sorting beans" to make sure the numbers match management's expectation, he says. Such detailed bean-counting, he says, has cut his direct involvement in inspections by 30 percent, leaving a larger proportion to less-experienced colleagues.

In addition, he says, inspecting by the numbers can misdirect efforts. "I have to put in a specific amount of time inspecting [plant] operations, even when I know the problems are in maintenance and engineering," he says.

What management really needs to know, he says, is whether conditions are improving or getting worse at a nuclear plant. "Instead, they are forcing senior resident inspectors to be gatekeepers of accounting stuff."

NRC administrators defend the procedures. Robert Gallo in the NRC's office of Nuclear Reactor Regulation, which runs the agency's licensing and inspection programs, notes that while the administrative workload is an issue among the senior resident inspectors, it's an inescapable part of the job. Part of that job is to train subordinates so that they become adept at ferreting out problems. "You can't do it all yourself," Mr. Gallo observes.

TURBINE TALK: The NRC's on-site inspectors are the agency's eyes and ears. Inspector Macdonald watches over the arrival of equipment in the turbine building at Seabrook.

When asked how tightly he has to hold to time allocations on the various inspections, Macdonald replies: "When I come across a safety issue, resolution is paramount, not accounting for time."

Critics outside the NRC add that too often some inspectors look only at a utility's documents to verify a plant is being well-run instead of checking hardware.

"Everyone starts with paper," acknowled_ one senior resident inspector at a plant in the South. "That's how you figure out the next step - where you need to crawl around and follow up." But he adds that some of his colleagues do stop at paperwork, adding, "It's a function of their energy level."

The problem with inspecting only paper, according to internal NRC reports, is that utilities often say they are correcting problems when they aren't.

WHEN a potential violation is found, it is turned over to regional NRC offices to pursue possible enforcement actions. But the response may vary, depending on the political climate in Washington, says Herb Livermore, who recently retired after more than a decade as a resident inspector. "If management got word from D.C. that we were getting too tough, you wouldn't get much support," he says. "If word came down that we were being too lax, you'd get more support."

A senior resident inspector adds: "The real problem is getting my [regional] management to dig in and hold the line" with headquarters.

At times, say some inspectors, they meet resistance ranging from stone-walling to harassment and intimidation.

Two years ago, regional reactor inspector Larry King and a colleague went to a nuclear plant near St. Lucie,

Fla., to follow up on items from previous inspections. What they found prompted his colleague to write up three violations, says Mr. King, who in 1994 won a harassment and intimidation action against the NRC after being denied more than a dozen promotions for persisting in pressing safety issues.

But the NRC section chief, who had been the senior resident inspector at the S . Lucie plant, "changed the violations to 'unresolved items,'" King says. He and his partner tried to voice their concerns in a larger performance assessment the NRC was conducting, "but our results were not included. Now they're finding all kind of problems at the plant."

Rebecca Long, an inspector in the NRC's Region 2 office in Atlanta, says that watering down reports without allowing inspectors to respond violates NRC policy. But, she says, King's experience isn't unique.

Ms. Long, who has won a sex and job discrimination case against the agency, had one supervisor who quietly withdrew a citation she had prepared for problems at a research reactor at the Georgia Institute of Technology in Atlanta. The reactor was allowed to operate until nearly a year later, when the agency shut it down following an accident that stemmed from the violations Long identified in her original citation.

Other supervisors made life even more difficult, Long says, after she found violations at the TVA's Browns Ferry and Watts Bar plants. Immediate supervisors were berating her work and downgrading her in job evaluations, she says, even as their superiors at regional and national headquarters were praising the quality of her inspection reports.

Today, Long carefully avoids describing anything that might constitute a violation of her settlement with the agency. But she notes that her victory has been bittersweet. "Nobody was ever punished," she says of her nine-year ordeal. "People did things the NRC manual says they should be terminated for."

The danger, critics say, is that such cases chill people who otherwise might raise safety concerns. "Many people have come to me and said that after seeing what I went through, they never would disagree with management," Long says. "They're afraid they'll get into trouble."

Tom Devine of the Government Accountability Project in Washington, which provides legal counsel for whistleblowers, agrees: "The NRC has the symptoms of an agency saturated with frustrated whistleblowers afraid to come out of the closet."

If Seabrook's Macdonald has been spared the frustrations some inspectors cite, he acknowledges the inherent pressures of the job. Inspectors often work 10- to 12-hour days. They must abide by regulations preventing them from socializing with utility workers.

The NRC's rotation policy can add to the sense of isolation: "We've moved four times since I joined the NRC. We've had to build protective walls to avoid being too deeply rooted in a community."

DUST-BUSTER: A worker at the Millstone plant in Waterford, Conn., vacuums up radioactive dust as a safety procedure. Major NRC violations can result in fines or plant closure.

"The resident program is not a career program," he adds. "Generally, you enter as a young person, take one, two, or three [plant] assignments, and move into a regional or headquarters job."

All of which raises another issue: experience. Being an on-site inspector is becoming "more and more of a young person's position," he says. The age and experience of the NRC's resident inspectors bear watching since these people are the ones who raise the majority of safety and performance issues, he says.

According to NRC figures, 75 percent of the agency's resident inspectors and 17 percent of the senior resident inspectors are in their first assignment. Thus, while they will have gone through a training and mentoring program, many have less than five years' experience in dealing directly with plants, utilities, and their own regional offices.

MACDONALD, who received the agency's Meritorious Service Award in 1993, says management changes have helped offset the trend. "Over the past six or seven years, the technical support [for inspectors] has improved dramatically," he says. "If I find an instrumentation issue and raise it during my early morning call to the regional office, I'll have a call back by 9 a.m. asking for details."

Some veteran inspectors, however, remain concerned that young, inexperienced watchdogs are unlikely to buck the system. "The senior resident who wants to be executive director of operations someday knows that sticking his neck out isn't going to do his career any good," says one veteran inspector in another region. "The older ones will tell you up front that to get along, you have to go along. They're starting to replace us old goats with lambs and sheep. That's where we're headed."

Miller ORP
JOM
Schedule attendance
by your staff
Luis

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

MEMORANDUM TO: Hubert J. Miller, Regional Administrator, Region I
Stewart D. Ebneter, Regional Administrator, Region II
A. B. Beach, Regional Administrator, Region III
Leonard J. Callan, Regional Administrator, Region IV
Steven A. Varga, Director, Division of Reactor Projects I/II
Jack W. Roe, Director, Division of Reactor Projects III/IV

FROM: M. Wayne Hodges, Director M. Wayne Hodges
Division of Systems Technology
Office of Nuclear Regulatory Research

SUBJECT: IPE SENIOR REVIEW BOARD SEPTEMBER MEETING AGENDA

Attached is the agenda for the September 11-12, 1996, IPE Senior Review Board meeting. The purpose of the meeting is to discuss the contractor-developed technical evaluation reports for St. Lucie and Commanche Peak and the staff evaluation report for Susquehanna. The meeting will be held in Two White Flint, room T-10-A-1.

Attached please find the detailed meeting agenda. As always, resident inspectors and project managers are invited to attend. If you have any questions, please contact John C. Lane at 301-415-6442.

2-1 OF 28

Attachment:
SRB September meeting agenda

cc: E. Butcher
L. Wiens
A. Gody, Jr. R-IV
A. Camp, SNL
J. Lehner, BNL
C. Poslusny
R. Hernan
M. Miller, R-II
T. Polich
J. Forrester, SNL
M. Banerjee, R-I

- Read IPE
- TER

EEE/138

The St. Lucie containment is a large, dry, steel containment vessel surrounded by an annular space and enclosed by a reinforced concrete shield building. The containment has a volume of approximately 2.5 million cu. ft. and a design pressure of 40 psig. The reactor coolant system is a Combustion Engineering (CE) two-loop design. Some of the plant characteristics important to the back-end analysis are summarized in Table 1 of this report.

Table 1: Plant and Containment Characteristics for St. Lucie Plant

Characteristic	St. Lucie	Zion	Surry
Thermal Power, MW(t)	2700	3236	2441
RCS Water Volume, ft^3	NP*	12,700	9200
Containment Free volume, ft^3	2,500,000	2,860,000	1,800,000
Mass of Fuel, lbm	207,000	216,000	175,000
Mass of Zircalloy, lbm	58,700	44,500	36,200
Containment Design Pressure, psig	40	47	45
Median Containment Failure Pressure, psig	95	135	126
RCS Water Volume/Power, $ft^3/MW(t)$	NP*	3.9	3.8
Containment Volume/Power, $ft^3/MW(t)$	926	884	737
Zr Mass/Containment Volume, lbm/ft^3	0.023	0.016	0.020
Fuel Mass/Containment Volume, lbm/ft^3	0.083	0.076	0.097

* Not provided in the IPE submittal.

The plant characteristics important to the back-end analysis are :

- A Steel containment that may be vulnerable to direct attack of dispersed core debris. However, according to the licensee's response to the RAI, the probability of the dispersed debris coming into contact with the containment steel shell is negligible because of the thermal shields around the vessel and the very narrow gap for the debris dispersion path.

- The large containment volume, high containment pressure capability, and the open nature of compartments which facilitates good atmospheric mixing.

- A cavity design which facilitates flooding of the reactor cavity. Ex-vessel cooling is likely to occur due to reactor cavity flooding and the low placement of the reactor vessel. This reduces the probability of vessel failure and is credited in CET quantification[2]. The cavity configuration is a deep cylinder, which would likely result in the formation of a deep molten core debris if all of the core mass pours into the cavity.

- There is no lower head penetrations in the St. Lucie reactor vessel. This may delay the time of vessel failure.

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[2] With the cavity flooded, a vessel failure probability of 0.1 (i.e., 0.9 probability of preventing vessel failure by ex-vessel cooling) is used in the IPE.

The Approach used for Back-End Analysis

The methodology employed in the St. Lucie IPE for the back-end evaluation is clearly described in the submittal. Containment event trees (CETs) were developed to determine the containment response and ultimately the type of release mode given that a core damage accident has occurred. The front-to-back end interface are provided in the IPE by the definition of 15 Plant Damage States (PDSs) for Unit 1 and 14 PDSs for Unit 2. These PDSs are identified by core damage state, determined by core melt timing and RCS pressure, and containment state, determined by containment pressure boundary status and containment safeguards system status.

The top events of the CET are quantified by the use of fault trees (called logic trees in the IPE submittal), which address the phenomenological, systems, and operator human response issues important to accident progression. The CET and the logic trees used in the St. Lucie IPE provide a structure for the evaluation of all of the containment failure modes discussed in NUREG-1335. The quantification of the CET in the St. Lucie IPE is based on NUREG-1150 data and plant-specific MAAP calculations The result of the Level 2 analysis are grouped to forty five release modes. Release fractions for these release modes are determined by the development of a parametric code similar to that used in NUREG-1150 (i.e., X-SOR) and plant-specific MAAP calculations.

For the St. Lucie Plant IPE, despite some inconsistencies, the definition of the interface between Level 1 and Level 2 analyses is in general reasonable. The CET is well structured and easy to understand. Although CET quantification and source term grouping and quantification seem adequate, the basis for some data used in CET quantification is not sufficiently discussed in the IPE submittal, and despite uncertainties of these data, their effect on CET quantification is not evaluated in the IPE (by sensitivity analyses).

Back-End Analysis Results

For St. Lucie, the leading PDS, which contributes about 18% to total CDF, is a PDS with early core melt, with the RCS at high pressure, and with all containment system available (PDS 3B). The accident sequences that contribute to this PDS are transient initiated sequences. This PDS is followed by a low pressure PDS (13%, PDS 5B) and an intermediate pressure PDS (12%, PDS 1B), both with early core melt and all containment system available, and another high pressure PDS with early melt but with no containment system unavailable (12%, PDS 3H). The latter high pressure PDS (PDS 3H) includes the SBO sequences and is the dominant contributor to both early and late containment failures.

Table E-3 shows the probabilities of containment failure modes for St. Lucie Plant as percentages of the total CDF. Results from the NUREG-1150 analyses for Surry and Zion are also presented for comparison.

Table E-3. Containment Failure as a Percentage of Total CDF

Containment Failure Mode	St. Lucie Plant IPE, Unit 1++	St. Lucie Plant IPE, Unit 2++	Surry NUREG-1150	Zion NUREG-1150
Early Failure	1	1	0.7	1.4
Late Failure	15	13	5.9	24.0
Bypass	12	15	12.2	0.7
Isolation Failure	***	***	*	**
Intact	72	71	81.2	73.0
CDF (1/ry)	2.3E-5	2.6E-5	4.0E-5	3.4E-4

++ The data presented for St. Lucie are based on Figure 4.0-4 of the IPE submittal. The difference between Unit 1 and Unit 2 is due to different Level 1 analysis results.

* Included in Early Failure, approximately 0.1%.

** Included in Early Failure, approximately 0.5%.

*** Included in Early Failure, 0.1%.

As shown in the above table, the conditional probability of containment bypass for St. Lucie is 12% of total CDF for Unit 1 and 15% for Unit 2. Containment bypass comes from ISLOCA and SGTR with ISLOCA being the primary contributor (70% of all bypass for Unit 1 and 77% for Unit 2).

The conditional probability of early containment failure for both Unit 1 and Unit 2 is about 1% of total CDF. According to the "Summary and Conclusions" section of the IPE submittal (Section 4.8), "The major contributors to early containment failure for St. Lucie include containment threats due to HPME loads from high RCS pressure core damage accidents, steam explosion events for low pressure sequences, and isolation failures." This is not really accurate. According to the results presented in the IPE submittal and the licensee's response to the RAI questions, early containment failure for St. Lucie is dominated by two CET end states (E3-R and E4-R) for PDS 3H. These two CET end states contribute over 70% of total early failure probability for St. Lucie, and both of them are associated with successful RCS depressurization (thus not from HPME), and with the major contributor to containment failure from overpressurization (with a conditional probability of 0.1), not steam explosion (with a conditional probability of 0.8%). HPME is not a major contributor because of the high probability of successful RCS depressurization. The inaccurate statement in the IPE submittal may indicate a lack of sufficient examination of the IPE results. Among the PDSs, early failure comes primarily from PDS 3H (high pressure PDSs, including SBO sequences, over 80% early failure probability). This is followed by PDS 2B (intermediate pressure PDS, primarily from small-small LOCA, about 10% early failure probability).

The conditional probability of late containment failure for St. Lucie is 15% of total CDF for Unit 1 and 13% for Unit 2. According to the "Summary and Conclusions" of the IPE (Section 4.8 of the submittal), "The major contributor to late containment failures is steam overpressure in long term (hydrogen burning is likely to be precluded due to the steam inerted containment atmosphere)." This is not completely accurate. It fails to mention that the major contributors to late containment failure are CET end states associated with core-concrete interaction (or coolable debris not formed ex-vessel). According to the data presented in the IPE, the probability of containment failure due to steam pressure alone (without CCI) is in general much less than that with CCI.

According to the results presented in the IPE submittal and the RAI response, late containment failure for St. Lucie is dominated by two CET end states, C4-L of PDS 3H and C5-L of PDS 2B. They contribute over 60% of total late containment failure probability for both units of St. Lucie. End State C4-L is associated with successful RCS depressurization, failure of in-vessel coolant recovery, and ex-vessel debris not cooled (i.e., with CCI). End State C5-L is associated with failure of RCS depressurization, failure of in-vessel coolant recovery, and ex-vessel debris not cooled. For all late failure probability, over 90% is due to overpressure failure associated with CCI. The contribution from steam pressurization alone is small.

PDS 3H is the major contributor to late containment failure (44% of all late failure for Unit 1, 43% for Unit 2). This is followed by PDS 2B (26% for Unit 1, 30% for Unit 2) and PDS 2F (16% for Unit 1 and 10% for Unit 2). The high late failure probability for these PDSs is partly due to the low in-vessel recovery probability of these PDSs[3].

Source terms are provided in the IPE for 45 release modes (i.e., the CET end states). The source terms for the release modes are calculated in the IPE using a combination of plant-specific MAAP calculations and the parametric model developed in NUREG-1150 (i.e., the X-SOR program). The approach seems appropriate and the use of plant-specific MAAP calculation results in the parametric model also seems reasonable. It is noted that, source terms calculated by the above method (with results presented in Table 4.0-7) are only for non-bypass release modes. Release fractions for bypass sequences can be obtained from the MAAP calculation results presented in the IPE submittal.

Sensitivity studies were performed in the St. Lucie IPE for MAAP calculations only. Although the CET quantification involves the use of assumptions and data that have significant uncertainties (e.g., the parameters that determine the probability of in-vessel recovery and ex-vessel cooling), the IPE does not provide a sensitivity study for CET quantification to evaluate the effects of these assumptions on the IPE results (e.g., containment failure probabilities).

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[3] In-vessel recovery precludes CCI and thus the challenge of late overpressure failure associated with CCI.

E.5 Vulnerabilities and Plant Improvements

Definition of vulnerability is provided in Section 2.3.3 of the IPE submittal. Vulnerability is not defined for Level 2, and no vulnerabilities were identified from Level 2 analysis. The back-end analysis did not identify the need for any plant improvements.

E.6 Observations

Although the licensee appears to have analyzed the design and operations of St. Lucie Plant to discover instances of particular vulnerability to core damage, there are some deficiencies in the IPE and the IPE submittal. The important points of the technical evaluation of the IPE back-end analysis are summarized below:

- The back-end portion of the IPE supplies a substantial amount of information with regards to the subject areas identified in Generic Letter 88-20.

- The St. Lucie Plant IPE provides an evaluation of all phenomena of importance to severe accident progression in accordance with Appendix I of the Generic Letter.

- The containment analyses indicate that the conditional probability of containment failure is 72% for Unit 1 and 71% for Unit 2. The conditional probability of containment bypass is about 12% for Unit 1 and 15% for Unit 2, the conditional probability of early containment failure is 1% for both Unit 1 and Unit 2, the conditional probability of isolation failure is about 0.1% for both units, and the conditional probability of late containment failure is 15% for Unit 1 and 13% for Unit 2.

- There are inconsistencies in the IPE submittal. Some of the inconsistencies involve the grouping of accident sequences to the plant damage states, and some are due to the use of the NSAC-60 study as the basis for the IPE submittal (i.e., modifications to the NSAC-60 study for the St. Lucie IPE are not reflected in the IPE submittal). According to the licensee's responses to the RAI questions, the inconsistencies are mostly on the conservative side and will not change the conclusions of the IPE.

- The probability of in-vessel recovery (i.e., core melt terminated) is high for most PDSs. This is due to the high probabilities of RCS depressurization, core coolant injection recovery, and core debris in a coolable formation assumed in the IPE. In-vessel recovery eliminates challenges to early containment failure due to HPME and reduces challenges to late containment failure associated with core concrete interaction. The effects of the assumptions related to in-vessel recovery on the overall containment failure probabilities for St. Lucie are not evaluated and discussed in the IPE submittal. Sensitivity studies of these assumptions are not performed in the St. Lucie IPE.

- The statement made in the "Summary and Conclusions" section of the IPE submittal (Section 4.8) that "The major contributors to early containment failure for St. Lucie include containment threats due to HPME loads from high RCS pressure core damage accidents, steam explosion events for low pressure sequences, and isolation failures." is not really accurate. According to the results presented in the IPE submittal and the licensee's response to the RAI questions early containment failure for St. Lucie is dominated by two CET end states for a high pressure PDS. These two CET end states are associated with successful RCS depressurization (thus not from HPME) and the major contributor to containment failure is from overpressurization (with a conditional probability of 0.1), not from steam explosion (with a conditional probability of 0.8%). HPME is not a major contributor because of high probability of successful RCS depressurization. The less-than-inaccurate statement in the IPE submittal may indicate a lack of sufficient examination of the IPE results.

  Similarly, the statement on late containment failure that "The major contributor to late containment failures is steam overpressure in long term (hydrogen burning is likely to be precluded due to the steam inerted containment atmosphere)" is not completely accurate. It fails to mention that the major contributors to late containment are CET end states associated with core-concrete interaction (or coolable debris not formed ex-vessel). According to the data presented in the IPE, the probability of containment failure due to steam pressure alone (without CCI) is in general much less than that with CCI.

- The licensee has addressed the recommendations of the CPI program.

Frequency

		FREQUENC	NVB		Late Failure																
	PDS	UNIT1	A1	A2	B1	B2-L	B2-R	B3-L	B3-R	B4-L	B4-R	B5-L	B5-R	B6-L	B6-R	C1-L	C1-R	C2-L	C2-R	C3-L	C3-R	C4-L
1	IB	2.76E-06	1.28E-09	2.57E-10	2.59E-12	1.12E-12				4.00E-12		1.37E-12				1.12E-08	2.82E-11	2.26E-09	5.66E-12	4.38E-08	1.10E-10	8.80E-09
2	IIA	1.73E-06	1.81E-09	7.94E-10	7.90E-12	2.60E-12		2.82E-11		9.26E-12		1.92E-12		6.31E-13		5.07E-09	1.27E-11	1.66E-09	4.17E-12	2.48E-08	6.22E-11	8.17E-09
3	IIB	1.54E-06										1.29E-08	3.24E-11	2.60E-09	6.51E-12							
4	IIE	1.49E-06	1.46E-08	2.93E-09	6.23E-10	1.25E-10	3.13E-13	3.90E-09		7.84E-10	1.97E-12	1.57E-10		3.14E-11		5.38E-09	1.35E-11	1.08E-09	2.71E-12	4.05E-08	1.02E-10	8.14E-09
5	IIF	9.31E-07	2.34E-07	7.69E-08	1.13E-08	3.69E-09	9.24E-12	3.87E-08	9.69E-11	1.27E-08	3.18E-11	1.23E-07	3.08E-10	4.04E-08	1.01E-10	1.45E-08	3.63E-11	4.76E-09	1.19E-11	5.05E-08	1.27E-10	1.66E-08
6	IIIB	3.93E-06	4.60E-09	9.24E-10	2.01E-11	4.05E-12				1.44E-11		2.10E-12		4.21E-13		1.29E-08	3.23E-11	2.59E-09	6.48E-12	6.31E-08	1.58E-10	1.27E-08
7	IIIH	2.72E-06	3.77E-08	4.29E-08	1.70E-10	1.93E-10	4.84E-13			1.78E-10	4.46E-13	3.99E-16						1.13E-07	2.83E-10			1.21E-06
8	VB	2.99E-06	2.38E-08	4.78E-09	1.02E-09	2.05E-10	5.14E-13	6.38E-09	1.60E-11	1.29E-09	3.23E-12	1.99E-09		4.04E-10	1.01E-12	1.12E-08	2.80E-11	2.25E-09	5.68E-12	7.11E-08	1.78E-10	1.44E-08
9	VIA	7.01E-07	4.22E-09	8.48E-10	1.84E-10	3.69E-11	9.25E-14	1.15E-09		2.32E-10	5.82E-13	3.61E-10		7.29E-11	1.82E-13	1.20E-09		1.68E-09	4.22E-12	8.39E-09	2.10E-11	1.18E-08
10	VIB	6.31E-07	5.05E-09	1.01E-09	2.16E-10	4.33E-11	1.09E-13	1.35E-09		2.72E-10	6.81E-13	4.24E-10		8.52E-11	2.13E-13	2.37E-09		4.77E-10	1.20E-12	1.51E-08	3.79E-11	3.04E-09
11	IR	2.87E-07																				
12	IIR	4.70E-07																				
13	IVB	[illegible]E-07																				
14	VIE	[illegible]22E-07																				
15	CB	1.74E-06																				
	TOTAL	2.26E-05	3.27E-07	1.31E-07	1.33E-08	4.30E-09	1.08E-11	3.15E-08	1.13E-10	1.54E-08	3.87E-11	1.39E-07	3.40E-10	4.36E-08	1.08E-10	6.38E-08	1.51E-10	1.30E-07	3.25E-10	3.17E-07	7.96E-10	1.29E-06

Fraction to total CDF

		Fraction	NVB		Late Failure																
	PDS	UNIT1	A1	A2	B1	B2-L	B2-R	B3-L	B3-R	B4-L	B4-R	B5-L	B5-R	B6-L	B6-R	C1-L	C1-R	C2-L	C2-R	C3-L	C3-R	C4-L
1	IB	12.22%	5.67E-05	1.14E-05	1.15E-07	4.96E-08				1.77E-07		6.07E-08				4.96E-04	1.25E-06	1.00E-04	2.51E-07	1.94E-03	4.87E-06	3.90E-04
2	IIA	7.66%	8.02E-05	2.63E-05	3.50E-07	1.15E-07		1.25E-06		4.10E-07		8.50E-08		2.80E-08		2.25E-04	5.62E-07	7.36E-05	1.85E-07	1.10E-03	2.75E-06	3.62E-04
3	IIB	6.82%										5.71E-04	1.43E-06	1.15E-04	2.88E-07							
4	IIE	6.60%	6.47E-04	1.30E-04	2.76E-05	5.54E-06	1.39E-08	1.73E-04		3.47E-05	8.71E-08	6.95E-06		1.39E-06		2.38E-04	5.98E-07	4.78E-05	1.20E-07	1.79E-03	4.52E-06	3.60E-04
5	IIF	4.12%	1.04E-02	3.41E-03	5.00E-04	1.63E-04	4.09E-07	1.71E-03	4.29E-06	5.61E-04	1.41E-06	5.45E-03	1.36E-05	1.79E-03	4.45E-06	6.42E-04	1.61E-06	2.11E-04	5.27E-07	2.24E-03	5.62E-06	7.34E-04
6	IIIB	17.40%	2.04E-04	4.09E-05	8.90E-07	1.79E-07				6.37E-07		9.30E-08		1.86E-08		5.71E-04	1.43E-06	1.15E-04	2.87E-07	2.79E-03	7.00E-06	5.62E-04
7	IIIH	12.05%	1.67E-03	1.90E-03	7.33E-06	8.55E-06	2.14E-08			7.88E-06	1.98E-08	1.77E-11						5.00E-03	1.25E-05			5.36E-02
8	VB	13.24%	1.05E-03	2.12E-04	4.52E-05	9.10E-06	2.28E-08	2.83E-04	7.09E-07	5.71E-05	1.43E-07	8.81E-05		1.79E-05	4.46E-08	4.96E-04	1.24E-06	9.98E-05	2.52E-07	3.15E-03	7.88E-06	6.37E-04
9	VIA	3.10%	1.87E-04	3.76E-05	8.15E-06	1.63E-06	4.10E-09	5.09E-05		1.03E-05	2.58E-08	1.60E-05		3.23E-06	8.07E-09	5.31E-05		7.45E-05	1.87E-07	3.72E-04	9.30E-07	5.22E-04
10	VIB	2.79%	2.24E-04	4.47E-05	9.57E-06	1.92E-06	4.83E-09	5.98E-05		1.20E-05	3.02E-08	1.88E-05		3.77E-06	9.45E-09	1.05E-04		2.11E-05	5.31E-08	6.69E-04	1.68E-06	1.35E-04
11	IR	1.27%																				
12	IIR	2.08%																				
13	IVB	1.05%																				
14	VIE	1.87%																				
15	CB	7.71%																				
	TOTAL	100.00%	1.45%	0.58%	0.06%	0.02%	0.00%	0.23%	0.00%	0.07%	0.00%	0.61%	0.00%	0.19%	0.00%	0.28%	0.00%	0.57%	0.00%	1.41%	0.00%	5.73%
			2.03%							1.19%											12.49%	

	Frequenc	Unit 1 Frequency									Early Failure						No Failure					
	PDS	C4-R	C5-L	C5-R	C6-L	C6-R	D1-R	D2-R	D3-R	D4-R	E1-R	E2-R	E3-R	E4-R	E5-R	E6-R	NCF	NCFA	NCFB	neg	TOTAL	Fraction
1	IB	2.21E-11	2.51E-09	6.30E-12	5.05E-10	1.27E-12	3.02E-11	1.11E-11		3.62E-12	1.96E-10		8.54E-10		6.16E-11		1.04E-09	2.58E-06	5.44E-08	4.93E-08	2.76E-06	100%
2	IIA	2.04E-11	1.41E-09	3.53E-12	4.64E-10	1.16E-12	1.83E-11	7.61E-12	1.06E-11	4.43E-12	1.23E-10		5.36E-10		6.87E-11		2.38E-09	1.62E-06	3.41E-08	3.09E-08	1.73E-06	100%
3	IIB		6.58E-07	1.65E-09	1.32E-07	3.31E-10			2.59E-09	9.47E-10					1.61E-08				7.12E-07	1.56E-12	1.54E-06	100%
4	IIE	2.04E-11	1.36E-09	3.40E-12	2.73E-10	6.82E-13	1.59E-11	5.83E-12		1.95E-12	1.03E-10		7.90E-10		3.32E-11		1.18E-09	1.34E-06	3.91E-08	2.66E-08	1.49E-06	100%
5	IIF	4.16E-11	1.34E-07	3.36E-10	4.40E-08	1.10E-10	3.86E-11	1.61E-11	1.01E-09	4.21E-10	2.60E-10		1.09E-09		6.53E-09		8.74E-14	3.41E-12	2.53E-12	1.15E-07	9.31E-07	100%
6	IIIB	3.18E-11	1.54E-09	3.86E-12	3.10E-10	7.74E-13	4.36E-11	1.59E-11		2.22E-12	2.82E-10		1.23E-09		3.78E-11		5.48E-09	3.72E-06	7.60E-08	3.03E-08	3.93E-06	100%
7	IIIH	3.02E-09			2.77E-13		1.19E-11	1.70E-11		3.32E-12	7.17E-09	7.21E-09	7.33E-08	7.37E-08	1.40E-09	1.40E-09	4.77E-09	1.12E-06	5.67E-10	2.35E-08	2.72E-06	100%
8	VB	3.62E-11	1.85E-08	4.64E-11	3.74E-09	9.39E-12	2.79E-11	1.03E-11	7.29E-11	2.69E-11	1.81E-10		1.39E-09		4.54E-10		1.30E-11	2.36E-06	8.51E-08	3.64E-07	2.97E-06	99.4%
9	VIA	2.95E-11	2.19E-09	5.48E-12	3.06E-09	7.71E-12	6.59E-12	2.41E-12	1.72E-11	6.29E-12	3.13E-11	1.14E-11	2.40E-10	8.76E-11	7.84E-11	2.87E-11	2.84E-10	5.58E-07	2.07E-08	8.58E-08	7.01E-07	100%
10	VIB	7.64E-12	3.91E-09	9.86E-12	7.89E-10	1.98E-12	5.93E-12	2.17E-12	1.55E-11	5.66E-12	3.85E-11		2.95E-10		9.64E-11		2.76E-12	5.01E-07	1.81E-08	7.72E-08	6.31E-07	100%
11	IR																					
12	IIR																					
13	IVB																					
14	VIE																					
15	CB																					
	TOTAL	3.23E-09	8.23E-07	2.06E-09	1.85E-07	4.64E-10	1.99E-10	8.84E-11	3.72E-09	1.42E-09	8.38E-09	7.22E-09	7.97E-08	7.38E-08	2.49E-08	1.43E-09	1.51E-08	1.38E-05	1.04E-06	8.03E-07		

	Fraction t	Unit 1 Frequency									Early Failure						No Failure				
	PDS	C4-R	C5-L	C5-R	C6-L	C6-R	D1-R	D2-R	D3-R	D4-R	E1-R	E2-R	E3-R	E4-R	E5-R	E6-R	NCF	NCFA	NCFB	neg	TOTAL
1	IB	9.79E-07	1.11E-04	2.79E-07	2.24E-05	5.62E-08	1.34E-06	4.91E-07		1.60E-07	8.68E-06		3.78E-05		2.73E-06		4.61E-05	1.14E-01	2.41E-03	2.18E-03	12.21%
2	IIA	9.04E-07	6.24E-05	1.56E-07	2.05E-05	3.14E-08	8.10E-07	3.37E-07	4.69E-07	1.96E-07	5.45E-06		2.37E-05		3.04E-06		1.05E-04	7.17E-02	1.51E-03	1.37E-03	7.67%
3	IIB		2.91E-02	7.31E-05	5.84E-03	1.47E-05			1.15E-04	4.19E-05					7.13E-04				3.15E-02	6.91E-08	6.82%
4	IIE	9.04E-07	6.02E-05	1.51E-07	1.21E-05	3.02E-08	7.04E-07	2.58E-07		8.64E-08	4.56E-06		3.50E-05		1.47E-06		5.23E-05	5.93E-02	1.73E-03	1.18E-03	6.59%
5	IIF	1.84E-06	5.93E-03	1.49E-05	1.95E-03	4.87E-06	1.71E-06	7.13E-07	4.47E-05	1.86E-05	1.15E-05		4.83E-05		2.89E-04		3.87E-09	1.51E-07	1.12E-07	5.09E-03	4.12%
6	IIIB	1.41E-06	6.82E-05	1.71E-07	1.37E-05	3.43E-08	1.93E-06	7.05E-07		9.83E-08	1.25E-05		5.45E-05		1.67E-06		2.43E-04	1.65E-01	3.37E-03	1.34E-03	17.42%
7	IIIH	1.34E-04			1.23E-08		5.27E-07	7.53E-07		1.47E-07	3.18E-04	3.19E-04	3.25E-03	3.26E-03	6.20E-05	6.20E-05	2.11E-04	4.96E-02	2.51E-05	1.04E-03	12.05%
8	VB	1.60E-06	8.19E-04	2.05E-06	1.66E-04	4.16E-07	1.24E-06	4.54E-07	3.23E-06	1.19E-06	8.02E-06		6.16E-05		2.01E-05		5.76E-07	1.05E-01	3.77E-03	1.61E-02	13.16%
9	VIA	1.31E-06	9.70E-05	2.43E-07	1.36E-04	3.41E-07	2.92E-07	1.07E-07	7.62E-07	2.78E-07	1.39E-06	5.06E-07	1.06E-05	3.88E-06	3.47E-06	1.27E-06	1.26E-05	2.47E-02	9.17E-04	3.80E-03	3.10%
10	VIB	3.38E-07	1.74E-04	4.37E-07	3.49E-05	8.77E-08	2.63E-07	9.61E-08	6.86E-07	2.51E-07	1.71E-06		1.31E-05		4.27E-06		1.22E-07	2.22E-02	8.02E-04	3.42E-03	2.79%
11	IR																				1.27%
12	IIR																				2.08%
13	IVB																				1.05%
14	VIE																				1.87%
15	CB																				7.71%
	TOTAL	0.01%	3.63%	0.01%	0.82%	0.00%	0.00%	0.00%	0.02%	0.01%	0.04%	0.03%	0.35%	0.33%	0.11%	0.01%	0.07%	61.11%	4.61%	3.55%	99.92%
							0.02%				0.87%						65.78%			3.55%	85.93%

Fre ency

	PDS	FREQUENC UNIT1	NVB A1	NVB A2	B1	B2-L	B2-R	B3-L	B3-R	B4-L	B4-R	B5-L	B5-R	B6-L	B6-R	Late Failure C1-L	C1-R	C2-L	C2-R	C3-L	C3-R	C4-L
1	IB	2.39E-06	1.11E-09	2.23E-10	2.24E-12	9.70E-13				3.47E-12		1.19E-12				9.70E-09	2.44E-11	1.96E-09	4.90E-12	3.79E-08	9.53E-11	7.62E-09
2	IIA	1.86E-06	1.95E-09	6.39E-10	8.49E-12	2.79E-12		3.03E-11		9.95E-12		2.06E-12		6.79E-13		5.45E-09	1.37E-11	1.79E-09	4.48E-12	2.67E-08	6.69E-11	8.78E-09
3	IIB	1.71E-06										1.43E-08	3.60E-11	2.89E-09	7.23E-12							
4	IIE	1.51E-06	1.48E-08	2.97E-09	6.31E-10	1.27E-10	3.17E-13	3.95E-09		7.94E-10	1.99E-12	1.59E-10		3.19E-11		5.45E-09	1.37E-11	1.09E-09	2.75E-12	4.10E-08	1.03E-10	8.24E-09
5	IIF	5.34E-07	1.34E-07	4.41E-08	6.48E-09	2.12E-09	5.30E-12	2.22E-08	5.56E-11	7.26E-09	1.83E-11	7.05E-08	1.77E-10	2.32E-08	5.77E-11	8.32E-09	2.08E-11	2.73E-09	6.83E-12	2.90E-08	7.28E-11	9.51E-09
6	IIIB	4.46E-06	5.22E-09	1.05E-09	2.28E-11	4.59E-12				1.63E-11		2.38E-12		4.77E-13		1.46E-08	3.67E-11	2.94E-09	7.36E-12	7.16E-08	1.79E-10	1.44E-08
7	IIIH	2.64E-06	3.66E-08	4.16E-08	1.65E-10	1.87E-10	4.70E-13			1.73E-10	4.33E-13	3.87E-16						1.10E-07	2.75E-10			1.17E-06
8	VB	2.86E-06	2.28E-08	4.57E-08	9.76E-10	1.96E-10	4.92E-13	6.10E-09	1.53E-11	1.23E-09	3.09E-12	1.90E-09		3.86E-10	9.64E-13	1.07E-08	2.68E-11	2.16E-09	5.43E-12	6.80E-08	1.70E-10	1.38E-08
9	VIA	7.18E-07	4.32E-09	8.69E-10	1.88E-10	3.78E-11	9.48E-14	1.18E-09		2.38E-10	5.96E-13	3.70E-10		7.47E-11	1.87E-13	1.23E-09		1.72E-09	4.32E-12	8.59E-09	2.15E-11	1.21E-08
10	VIB	1.08E-06	8.64E-09	1.73E-09	3.70E-10	7.41E-11	1.87E-13	2.31E-09		4.65E-10	1.17E-12	7.26E-10		1.46E-10	3.65E-13	4.06E-09		8.16E-10	2.05E-12	2.58E-08	6.49E-11	5.21E-09
11	IR																					
12	IIR	7.94E-07																				
13	IVB	1.32E-07																				
14	VIE	4.25E-07																				
15	CB	2.72E-06																				
	TOTAL	2.38E-05	2.30E-07	1.39E-07	[illegible]5E-09	2.75E-09	6.86E-12	3.58E-08	7.09E-11	1.02E-08	2.53E-11	8.80E-08	2.13E-10	2.67E-08	6.64E-11	5.96E-08	1.36E-10	1.23E-07	3.13E-10	3.09E-07	7.74E-10	1.25E-06

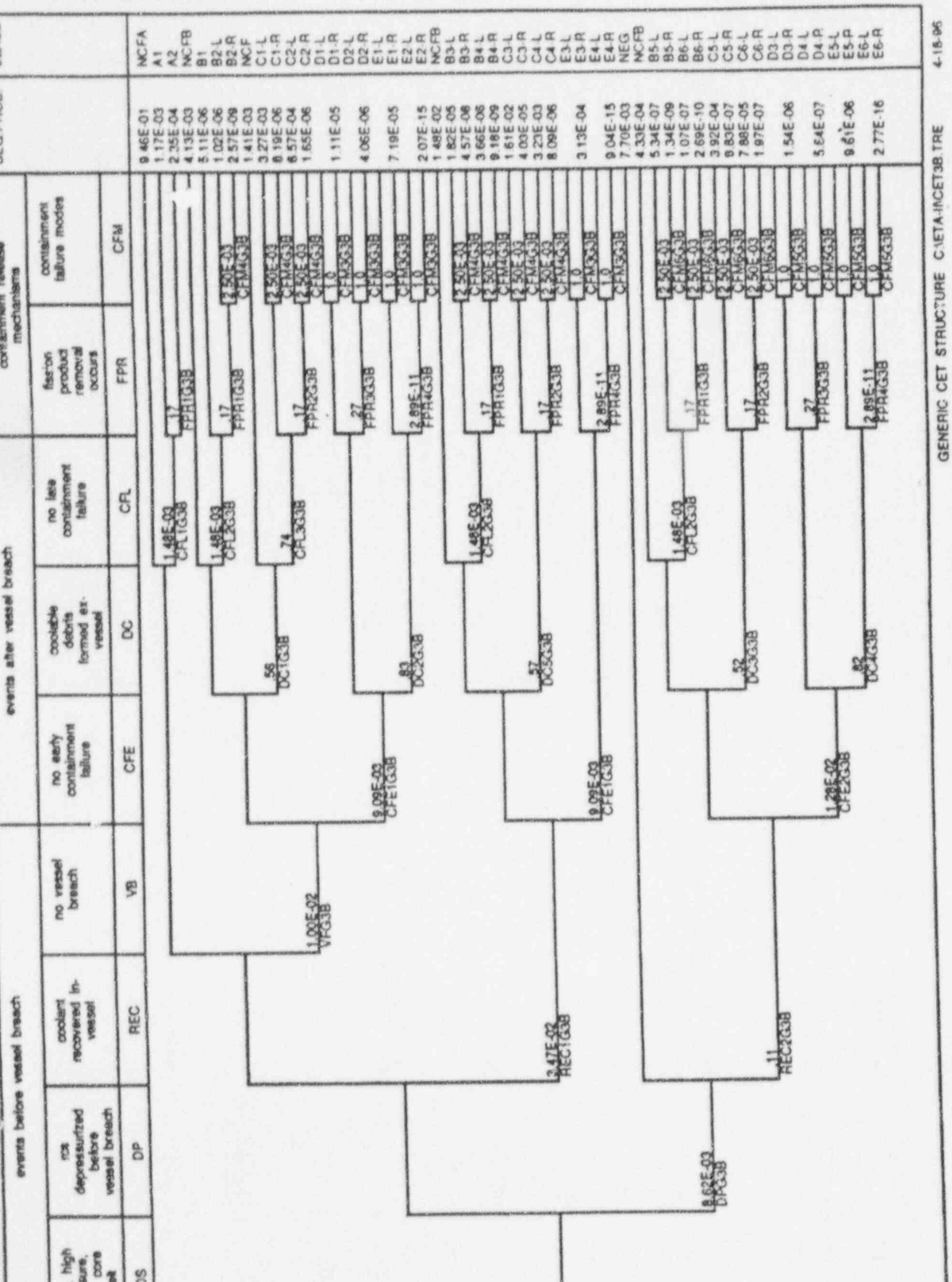
Fraction to total CDF

	PDS	Fraction UNIT2	NVB A1	NVB A2	B1	B2-L	B2-R	B3-L	B3-R	B4-L	B4-R	B5-L	B5-R	B6-L	B6-R	Late Failure C1-L	C1-R	C2-L	C2-R	C3-L	C3-R	C4-L
1	IB	10.03%	4.65E-05	9.34E-06	9.41E-08	4.07E-08				1.45E-07		4.98E-08				4.07E-04	1.02E-06	8.21E-05	2.06E-07	1.59E-03	4.00E-06	3.20E-04
2	IIA	7.80%	8.17E-05	2.68E-05	3.56E-07	1.17E-07		1.27E-06		4.18E-07		8.66E-08		2.85E-08		2.29E-04	5.73E-07	7.49E-05	1.88E-07	1.12E-03	2.81E-06	3.68E-04
3	IIB	7.17%										6.01E-04	1.51E-06	1.21E-04	3.03E-07							
4	IIE	6.34%	6.21E-04	1.25E-04	2.65E-05	5.32E-06	1.33E-08	1.66E-04		3.33E-05	8.36E-08	6.68E-06		1.34E-06		2.29E-04	5.74E-07	4.59E-05	1.15E-07	1.72E-03	4.34E-06	3.46E-04
5	IIF	2.24%	5.63E-03	1.85E-03	2.72E-04	8.88E-05	2.22E-07	9.31E-04	2.33E-06	3.05E-04	7.66E-07	2.96E-03	7.41E-06	9.72E-04	2.42E-06	3.49E-04	8.74E-07	1.15E-04	2.86E-07	1.22E-03	3.06E-06	3.99E-04
6	IIIB	18.71%	2.19E-04	4.40E-05	9.57E-07	1.93E-07				6.85E-07		1.00E-07		2.00E-08		6.14E-04	1.54E-06	1.23E-04	3.09E-07	3.00E-03	7.52E-06	6.04E-04
7	IIIH	11.08%	1.54E-03	1.75E-03	6.92E-06	7.86E-06	1.97E-08			7.24E-06	1.82E-08	1.62E-11						4.60E-03	1.15E-05			4.93E-02
8	VB	12.00%	9.55E-04	1.92E-03	4.09E-05	8.24E-06	2.06E-08	2.56E-04	6.42E-07	5.17E-05	1.30E-07	7.99E-05		1.62E-05	4.04E-08	4.50E-04	1.12E-06	9.05E-05	2.28E-07	2.85E-03	7.14E-06	5.77E-04
9	VIA	3.01%	1.81E-04	3.64E-05	7.91E-06	1.58E-06	3.98E-09	4.94E-05		9.97E-06	2.50E-08	1.55E-05		3.13E-06	7.83E-09	5.16E-05		7.23E-05	1.81E-07	3.61E-04	9.02E-07	5.06E-04
10	VIB	4.53%	3.63E-04	7.25E-05	1.55E-05	3.11E-06	7.84E-09	9.70E-05		1.95E-05	4.89E-08	3.04E-05		6.12E-06	1.53E-08	1.70E-04		3.43E-05	8.61E-08	1.08E-03	2.72E-06	2.18E-04
11	IR																					
12	IIR	3.33%																				
13	IVB	0.55%																				
14	VIE	1.78%																				
15	CB	11.41%																				
	TOTAL	100.00%	0.96%	0.58%	0.04%	0.01%	0.00%	0.15%	0.00%	0.04%	0.00%	0.37%	0.00%	0.11%	0.00%	0.25%	0.00%	0.52%	0.00%	1.30%	0.00%	5.26%
			1.55%							0.72%											11.65%	

	Frequenc	Unit 1 Frequency																				
							Early Failure										No Failure					
	PDS	C4-R	C5-L	C5-R	C6-L	C6-R	D1-R	D2-R	D3-R	D4-R	E1-R	E2-R	E3-R	E4-R	E5-R	E6-R	NCF	NCFA	NCFB	neg	TOTAL	Fraction
1	IB	1.91E-11	2.17E-09	5.46E-12	4.37E-10	1.10E-12	2.62E-11	9.61E-12		3.13E-12	1.70E-10		7.40E-10		5.33E-11		9.01E-10	2.23E-06	4.71E-08	4.27E-08	2.39E-06	100%
2	IIA	2.19E-11	1.52E-09	3.80E-12	4.98E-10	1.25E-12	1.97E-11	8.18E-12	1.14E-11	4.76E-12	1.32E-10		5.76E-10		7.39E-11		2.56E-09	1.74E-06	3.67E-08	3.32E-08	1.86E-06	100%
3	IIB		7.31E-07	1.83E-09	1.47E-07	3.68E-10			2.88E-09	1.05E-09					1.79E-08				7.91E-07	1.73E-12	1.71E-06	100%
4	IIE	2.07E-11	1.38E-09	3.45E-12	2.76E-10	6.92E-13	1.61E-11	5.90E-12		1.98E-12	1.04E-10		8.01E-10		3.36E-11		1.20E-09	1.36E-06	3.96E-08	2.70E-08	1.51E-06	100%
5	IIF	2.39E-11	7.69E-08	1.93E-10	2.53E-08	6.30E-11	2.21E-11	9.23E-12	5.79E-10	2.41E-10	1.49E-10		6.25E-10		3.75E-09		5.01E-14	1.96E-12	1.45E-12	6.60E-08	5.34E-07	100%
6	IIIB	3.61E-11	1.75E-09	4.38E-12	3.52E-10	8.79E-13	4.95E-11	1.81E-11		2.52E-12	3.20E-10		1.40E-09		4.29E-11		6.22E-09	4.22E-06	8.62E-08	3.44E-06	4.46E-06	100%
7	IIIH	2.93E-09			2.69E-13		1.16E-11	1.65E-11		3.22E-12	6.96E-09	7.00E-09	7.11E-08	7.15E-08	1.36E-09	1.36E-09	4.63E-09	1.09E-06	5.50E-10	2.28E-08	2.64E-06	100%
8	VB	3.46E-11	1.77E-08	4.44E-11	3.58E-09	8.98E-12	2.67E-11	9.81E-12	6.97E-11	2.57E-11	1.73E-10		1.33E-09		4.34E-10		1.24E-11	2.26E-06	8.14E-08	3.48E-07	2.88E-06	101%
9	VIA	3.02E-11	2.24E-09	5.61E-12	3.14E-09	7.90E-12	6.75E-12	2.47E-12	1.76E-11	6.44E-12	3.21E-11	1.17E-11	2.46E-10	8.98E-11	8.03E-11	2.94E-11	2.91E-10	5.72E-07	2.12E-08	8.79E-08	7.18E-07	100%
10	VIB	1.31E-11	6.73E-09	1.69E-11	1.35E-09	3.39E-12	1.01E-11	3.72E-12	2.65E-11	9.69E-12	6.59E-11		5.05E-10		1.65E-10		4.72E-12	8.57E-07	3.10E-08	1.32E-07	1.08E-06	100%
11	IR																					
12	IIR																					
13	IVB																					
14	VIE																					
15	CB																					
	TOTAL	3.13E-09	8.41E-07	2.11E-09	1.81E-07	4.55E-10	1.89E-10	8.35E-11	3.58E-09	1.33E-09	8.11E-09	7.01E-09	7.74E-08	7.16E-08	2.39E-08	1.39E-09	1.58E-08	1.43E-05	1.13E-06	7.94E-07		

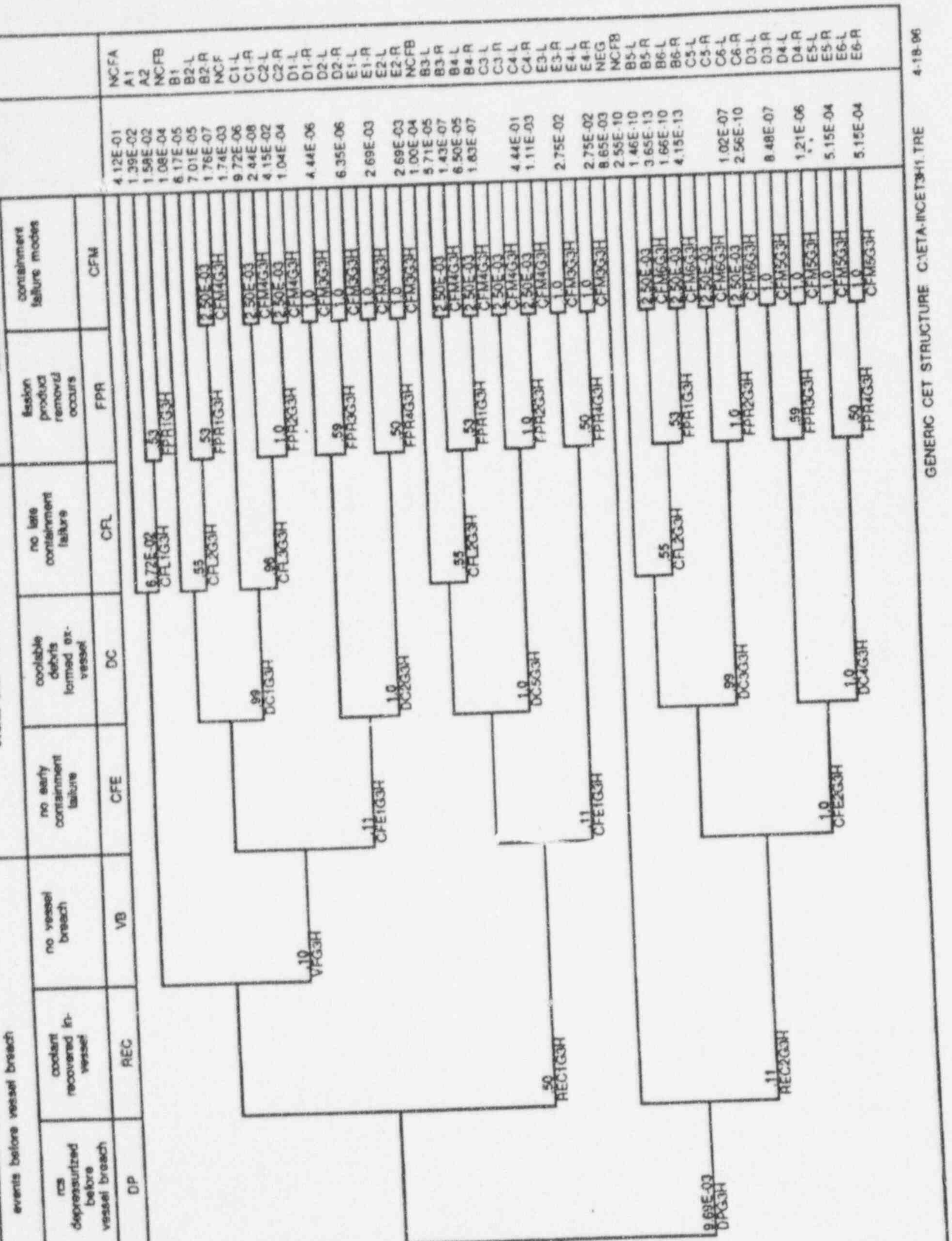
	Fraction 1	Unit 1 Frequency																			
							Early Failure										No Failure				
	PDS	C4-R	C5-L	C5-R	C6-L	C6-R	D1-R	D2-R	D3-R	D4-R	E1-R	E2-R	E3-R	E4-R	E5-R	E6-R	NCF	NCFA	NCFB	neg	TOTAL
1	IB	8.03E-07	9.12E-05	2.29E-07	1.84E-05	4.61E-08	1.10E-06	4.03E-07		1.31E-07	7.12E-06		3.10E-05		2.24E-06		3.78E-05	9.37E-02	1.98E-03	1.79E-03	10.02%
2	IIA	9.21E-07	6.36E-05	1.59E-07	2.09E-05	5.24E-08	8.26E-07	3.43E-07	4.78E-07	2.00E-07	5.55E-06		2.42E-05		3.10E-06		1.07E-04	7.31E-02	1.54E-03	1.39E-03	7.81%
3	IIB		3.07E-02	7.69E-05	6.15E-03	1.54E-05			1.21E-04	4.41E-05					7.50E-04				3.32E-02	7.27E-08	7.17%
4	IIE	8.68E-07	5.78E-05	1.45E-07	1.16E-05	2.90E-08	6.76E-07	2.48E-07		8.30E-08	4.38E-06		3.36E-05		1.41E-06		5.02E-05	5.70E-02	1.66E-03	1.13E-03	6.33%
5	IIF	1.00E-06	3.22E-03	8.09E-06	1.06E-03	2.64E-06	9.29E-07	3.87E-07	2.43E-05	1.01E-05	6.26E-06		2.62E-05		1.57E-04		2.10E-09	8.21E-08	6.09E-08	2.77E-03	2.24%
6	IIIB	1.51E-06	7.33E-05	1.84E-07	1.48E-05	3.69E-08	2.08E-06	7.58E-07		1.06E-07	1.34E-05		5.86E-05		1.80E-06		2.61E-04	1.77E-01	3.62E-03	1.44E-03	18.72%
7	IIIH	1.23E-04			1.13E-08		4.85E-07	6.92E-07		1.35E-07	2.92E-04	2.94E-04	2.99E-03	3.00E-03	5.70E-05	5.70E-05	1.94E-04	4.56E-02	2.31E-05	9.57E-04	11.08%
8	VB	1.45E-06	7.42E-04	1.86E-06	1.50E-04	3.77E-07	1.12E-06	4.12E-07	2.93E-06	1.08E-06	7.26E-06		5.58E-05		1.82E-05		5.22E-07	9.47E-02	3.42E-03	1.46E-02	12.10%
9	VIA	1.27E-06	9.41E-05	2.36E-07	1.32E-04	3.31E-07	2.83E-07	1.04E-07	7.39E-07	2.70E-07	1.35E-06	4.91E-07	1.03E-05	3.77E-06	3.37E-06	1.23E-06	1.22E-05	2.40E-02	8.90E-04	3.69E-03	3.01%
10	VIB	5.48E-07	2.82E-04	7.08E-07	5.66E-05	1.42E-07	4.26E-07	1.56E-07	1.11E-06	4.06E-07	2.76E-06		2.12E-05		6.92E-06		1.98E-07	3.60E-02	1.30E-03	5.54E-03	4.53%
11	IR																				3.33%
12	IIR																				0.55%
13	IVB																				1.78%
14	VIE																				11.41%
15	CB																				
	TOTAL	0.01%	3.53%	0.01%	0.76%	0.00%	0.00%	0.00%	0.02%	0.01%	0.03%	0.03%	0.32%	0.30%	0.10%	0.01%	0.07%	60.12%	4.76%	3.33%	100.10%
							0.02%				0.79%						64.95%			3.33%	83.02%

Attachment 13



PDS3B

Attachment 13



PDS3H

ST. LUCIE 1&2 BACK-END PRESENTATION

I.1 Contractor Observations & Conclusions:

Adequate but poor back-end analysis. CET quantification based on NUREG-1150 data, scoping values, and plant-specific MAAP calculations. However, sufficient discussions are not provided in the IPE submittal for the values used for some CET basic events that may have significant uncertainties and significant effect on CET results (e.g., parameters associated with in-vessel recovery). Level 2 results do not seem to be thoroughly examined.

I.2 Overall Reject or Accept Submittal: **Accept with recommendations.**

II. RAI Evaluation:

II.1 Adequacy of Responses: **Brief, some part of questions not specifically addressed. Adequate but poor.**

II.2 Remaining Concerns: **None, but see I.1 above.**

III Plant Characterization:

III.1 What Drives Conditional containment Failure Probability: **Early release is dominated by containment bypass (12% for Unit 1, 15% for Unit 2, primarily ISLOCA); Early containment failure (1%) is dominated by overpressure failure at low pressure VB; Late failure is dominated by CCI events.**

III.2 Unique Features/Issues Associated with the Plant: **A reactor cavity design that facilitates flooding and a low placement of the reactor vessel. This increases the probability of ex-vessel cooling. A steel containment. No lower head vessel penetrations.**

III.3 Comparisons with Similar Plants/Containment Types and 1150: **Both the thermal power and containment volume are between those of Zion and Surry.**

ST. LUCIE 1&2 BACK-END PRESENTATION

IV Licensee's IPE Review Process

IV.1 Completeness Assessment: **Complete.**

IV.2 Methodology Assessment/Were All Important Phenomenology Considered? **Yes. Using CET with fault trees for the quantification of CET top events.**

IV.3 Concerns Regarding Multi-Unit Effects/As-Built Conditions Different from IPE: **None.**

IV.4 Adequacy of Licensee Participation and Peer Review: **Adequate.**

IV.5 Did the Licensee Perform Sensitivity studies? **For MAAP analyses only. Not for CET quantification. Similar to many other IPEs, but the assumptions used in the St. Lucie IPE may have a significant effect on CET results.**

V. Containment analysis Assessment:

V.1 Conditional Failure Probabilities for Early failure, bypass Failure, and Late Failure Including Main Contributors to Each:

Type Failure	Failure Probability (%)	Contributors
Early	1	Transient(includ. SBO)
Bypass	12% for Unit 1 15% for Unit 2	ISLOCA (over 70%) & SGTR (about 30%)
Isolation	0.1	Independent fault tree analysis
Late	15% for Unit 1 13% for Unit 2	Transient (includ. SBO) & Small-small LOCA
Intact	72% for Unit 1 71% for Unit 2	

V.2 Mapping of Accident Sequences to Containment Failure Modes: **By 15 PDSs for Unit 1 and 14 PDSs for Unit 2 and CETs with 8 top events.**

V.3 Adequacy of Estimate for Containment Isolation Failure Probability: **Adequate.**

V.4 Vulnerabilities Identified/Plant Improvements Identified: **None.**

ST. LUCIE 1&2 BACK-END PRESENTATION

VI. Accident Progression and Containment Behavior Assessment

VI.1 Important Modeling Assumptions: **High probabilities of successful RCS depressurization by operator actions, recovery of coolant injection, and in-vessel recovery for most PDSs.**

VI.2 Impact of Changes Made on Containment Performance: **None.**

VI.3 Any Credit Taken for Operator Recovery Actions? **RCS depressurization, power recovery, recovery of coolant injection.**

VI.4 Codes Used for Analyzing Accident Progression and Source Terms: **A parametric code similar to that used in NUREG-1150 and MAAP code for plant specific calculations.**

VII. Fission Product Release Assessment

VII.1 Adequacy in Relation to IPE Reporting Criteria: **Adequate.**

VII.2 Major Sequences Contributing to Releases: **Bypass sequences.**

VIII. CPI Program Assessment:

VIII.1 Were global burns analyzed? **Yes.**

VIII.2 Were walkdowns performed to address local hydrogen pocketing? **Yes.**

VIII.3 Was equipment survivability addressed including recovery of failed equipment? **Although there were basic events in the CET structure that seem to address this issue, the RAI response stated that the survivability issue was not considered in the CET explicitly, but was embodied in the MAAP analysis.**

VIII.4 Were high temperature effects on seal considered? **Yes, qualitatively (in the RAI response).**

Front-End TER Presentation for St. Lucie

Observations and Conclusions	Strengths: (1) Comprehensive treatment of plant specific initiating events; (2) usage of plant specific data where possible; common cause factors are generally OK; (3) results make sense and insights derived; (4) system descriptions; (5) utility involvement; (6) ABAO plant modeling. Weaknesses were mostly in data, paucity of PS plant specific categories and CCF categories, RAI Request Additional Info. discussions, ATWS modeling. Overall Assessment of IPE: About average Recommendation: Accept IPE
RAI Evaluation	Uneven
Plant Characterization	CDF Results comparable to other typical PWRs. Largest class of accidents contributing to CDF is LOCAs. Important contributors are various failures of HPI, including support system (CCW). SBO contributes about 10%. Many crossties between units possible (not sure which ones credited), 8 hour battery life (only 4 hours credited), TD pump control post battery depletion not credited, self-cooled EDGs, power recovery factors optimistic (factor of 2-3), IE frequency seems more realistic than at other plants (0.15/yr). if you shed load w/in 1hr Other unique features: (1) MFW pumps will continue running after most transients, (2) high level of redundancy of instrument air system, (3) cross connect of CST, IA, EPS, (4) HVAC relatively unimportant, (5) Byrin-Jackson RCP seals cooled by CCW only (injection disconnected), requiring 10 min operator action upon CCW loss, (6) U2 operates with one block valve closed; (7) recirc switchover automatic (8) open plant; (9) condensate system (3 pumps) can be used < 600 psi for SG cooling.
Licensee's IPE Process	Small event tree/large fault tree; used CAFTA. Uncertainty, importance (incl. systems) and sensitivity analyses provided.

CDF 2.6E-5

PORV capacity too high. (LTOP considerations)

Front-End TER Presentation for St. Lucie

Accident Sequence Delineation and Systems Analysis PORVs Δ [illegible] between units	25 initiating events: 4 LOCAs, 12 transients, 9 support systems; additionally ISLOCA, ATWS, several flooding initiators (5); frequencies generally comparable to other IPE/PRAs. Success criteria based on CE non-proprietary reports, FSAR, and plant-specific MAAP analyses performed for IPE. Containment cooling considered in front-end analysis needed to support core cooling for most initiators, depending on sequence. EQ effects were also considered for affected sequences. Dependencies appear to have been suitably accounted for.
Quantitative Process Frequency to [illegible]	Analysis used functional event trees. Event tree sequence cutsets were quantified using a truncation value of 1.E-6 to 1.E-8, without IE (i.e., set to 1) and prior to recovery actions. Mean values were used for point estimate frequencies and probabilities. Recovery actions were considered. Uncertainty, importance and sensitivity analyses were provided and discussed in the Submittal and RAI Responses. Plant experience 1/85 to 10/91 (U1) and 6/86 to 4/92 (U2) used to quantify component failure rates. Plant specific failure data comparable to values used in other IPEs/PRAs. Types of pumps, valves not distinguished according to service (e.g., salt water vs. fresh water). Generic data are comparable to generic data used in most IPEs/PRAs (TDAFW pump?). Many components use generic data. The beta factor method used for common cause analysis; data source was SAIC generic CC database. Components considered for CCF comparable to those used in other IPEs/PRAs, with some omissions (all 3 AFW pumps, ckt breakers, relays, inverters, switches, transmitters, solenoid valves). CCF factors generally OK.

Front-End TER Presentation for St. Lucie

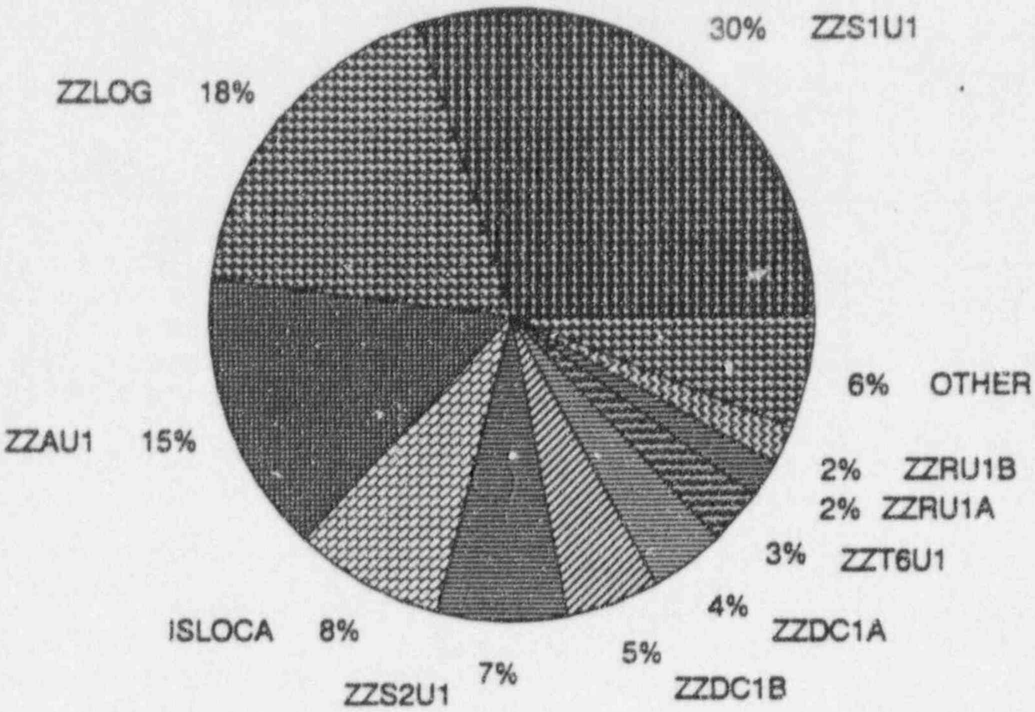
Interface Issues	Level 1 binning of core damage sequences into classes of accidents was provided. Important recovery actions considered include recovery of offsite power (optimistic). Containment cooling/core damage interface evaluated.
Evaluation of DHR and Other Safety Issues	Submittal discussion of DHR focuses on secondary cooling (MFW, condensate and AFW) and once through cooling (feed and bleed). CCW used to cool HPI, CS and SDC systems. AFW located outdoors, does not need HVAC. EDGs are self cooled. RAI responses briefly discussed insights into broader aspects of DHR. No vulnerabilities associated with DHR were identified. Other USIs/GSIs addressed in the IPE are: none (reserve the right to add in the future).
Internal Flooding	Analysis of internal flooding OK. The most important contributors identified. Five of the flooding scenarios contribute a dominant sequence.
Core Damage Sequence Results	CDF results - see attached tables. Systems and operator actions with greatest contribution to CDF are identified. Licensee defined a vulnerability as a sequence causing a disproportionately high CDF contribution, or early release contribution. Based on these criteria, no vulnerabilities were explicitly identified. One minor procedural improvement has been implemented.

HPI pumps have 1 common suction line

?. Isolation of non-safety portion of CCW is auto action

* TER needs a statement about why LOCA contribution is so high. Should be [illegible].
(No credit for recovery actions??)

Figure 3.7-2 St. Lucie Unit 1 Core Damage Frequency by Initiator

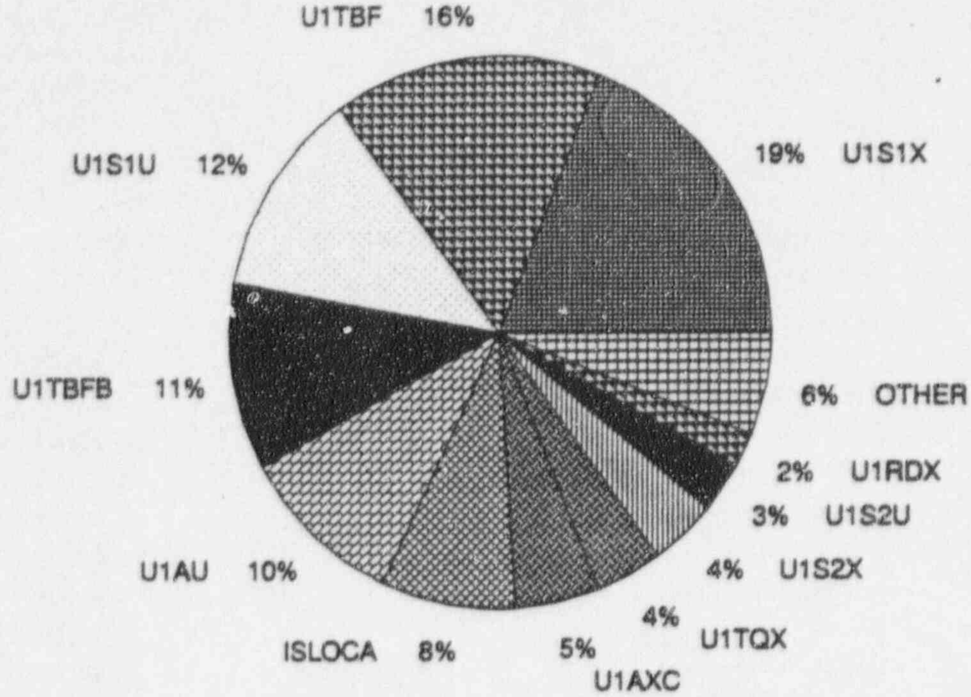


Initiator	Description	Core Damage Frequency Contribution	% of Total
ZZS1U1	Small-Small LOCA	7.09E-06	30%
ZZLOG	Loss of Grid	4.11E-06	18%
ZZAU1	Large LOCA	3.48E-06	15%
ISLOCA	Interfacing System LOCA	1.74E-06	8%
ZZS2U1	Small LOCA	1.60E-06	7%
ZZDC1B	Loss of DC Bus 1B	1.08E-06	5%
ZZDC1A	Loss of DC Bus 1A	9.75E-07	4%
ZZT6U1	Steamline Break Downstream of MSIVs	6.60E-07	3%
ZZRU1A	SGTR - S/G 1A	4.29E-07	2%
ZZRU1B	SGTR - S/B 1B	3.86E-07	2%
Other		1.59E-06	6%
Total Freq:		2.3E-05	

Table 3.7-2 St. Lucie Unit 1 Core Damage Frequency by Initiator

Initiator	Description	Core Damage Frequency Contribution	% of Total
ZZS1U1	Small-Small LOCA	7.10E-06	31%
ZZLOG	Loss of Grid	4.08E-06	18%
ZZAU1	Large LOCA	3.49E-06	15%
ISLOCA	Interfacing System LOCA	1.74E-06	8%
ZZS2U1	Small LOCA	1.60E-06	7%
ZZDC1B	Loss of DC Bus 1B	1.10E-06	5%
ZZDC1A	Loss of DC Bus 1A	9.90E-07	4%
ZZT6U1	Steamline Break Downstream of MSIVs	6.60E-07	3%
ZZRU1A	SGTR - S/G 1A	4.35E-07	2%
ZZRU1B	SGTR - S/G 1B	3.90E-07	2%
ZZCCWU1	Loss of CCW	2.83E-07	1%
ZZT3CU1	LOFW - Not Recoverable	2.38E-07	1%
ZZT7SIU1	Spurious SIAS	1.91E-07	1%
ZZT1U1	Reactor Trip	1.80E-07	1%
ZZT3AU1	LOFW - Recoverable	9.95E-08	<1%
ZZT2U1	Reactor Trip (PORV Challenge)	9.49E-08	<1%
ZZT5U1A	Upstream Steamline Break - S/G 1A	8.90E-08	<1%
ZZT5U1B	Upstream Steamline Break - S/G 1B	8.55E-08	<1%
ZZICWU1	Loss of ICW	8.06E-08	<1%
ZZT3EU1	Excessive Feedwater	6.82E-08	<1%
ZZIAU1	Loss of Instrument Air	6.08E-08	<1%
ZZT8AU1	PORV Sticking Open - S/G 1A	3.75E-08	<1%
ZZT8BU1	PORV Sticking Open - S/G 1B	3.74E-08	<1%
ZZT7MSU1	Spurious Main Steam Isolation	2.31E-08	<1%
ZZMAU1	Loss of Instrument Bus 1MA	4.39E-09	<1%
ZZMBU1	Loss of Instrument Bus 1MB	4.39E-09	<1%
ZZMCU1	Loss of Instrument Bus 1MC	4.34E-09	<1%
ZZMDU1	Loss of Instrument Bus 1MD	4.34E-09	<1%
ZZ4KV1B2	Loss of 4kV Bus 1B2	8.35E-10	<1%
ZZ4KV1A2	Loss of 4kV Bus 1A2	4.05E-10	<1%
ZZT3DU1A	Feedline Break S/G 1A	2.81E-10	<1%
ZZT3DU1B	Feedline Break S/G 1B	2.72E-10	<1%
ZZTCWU1	Loss of TCW	2.50E-10	<1%
ZZT3DU1	Feedline Break (Common)	2.34E-10	<1%
ZZ6KV1A1	Loss of 6.9kV Bus 1A1	9.54E-11	<1%
ZZ6KV1B1	Loss of 6.9kV Bus 1B1	9.54E-11	<1%
ZZT4B	Loss of Offsite Power "B" Train	6.38E-12	<1%
ZZT4A	Loss of Offsite Power "A" Train	4.38E-12	<1%
Total Freq:		2.32E-05	

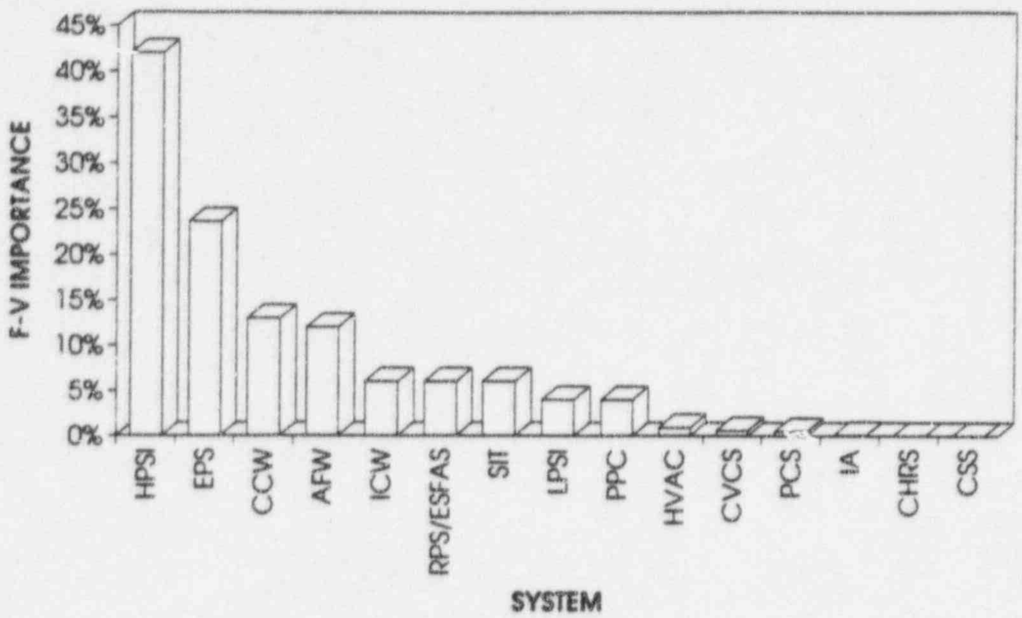
Figure 3.7-1 St. Lucie Unit 1 Core Damage Frequency by Sequence



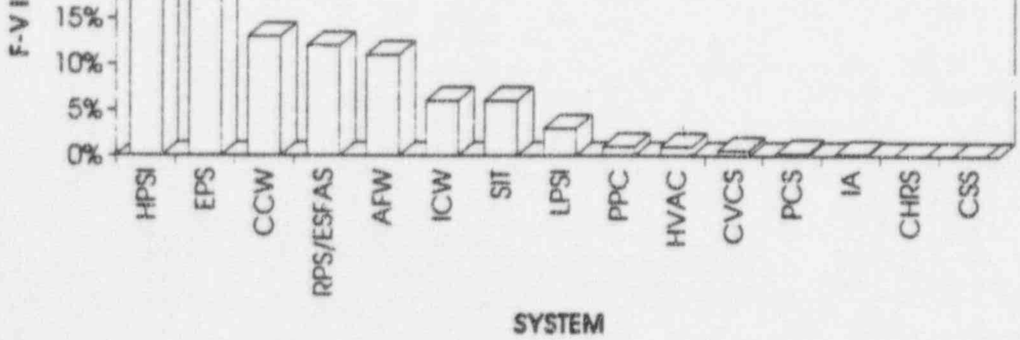
Sequence	Description	Core Damage Frequency Contribution	% of Total
U1S1X	Small-Small LOCA w/Long Term Core Cooling Failure	4.34E-06	19%
U1TBF	Transient w/Secondary Heat Removal and OTC Failure (Non-Blackout)	3.72E-06	16%
U1S1U	Small-Small LOCA w/Injection Failure	2.74E-06	12%
U1TBFB	Transient w/Secondary Heat Removal and OTC Failure (Blackout)	2.64E-06	11%
U1AU	Large LOCA w/Injection Failure	2.37E-06	10%
ISLOCA	Interfacing System LOCA	1.74E-06	8%
U1AXC	Large LOCA w/Cold Leg Recirc Failure	1.12E-06	5%
U1TQX	Transient w/Loss of RCS Integrity and Long Term Core Cooling Failure	9.31E-07	4%
U1S2X	Small LOCA w/Long Term Core Cooling Failure	9.29E-07	4%
U1S2U	Small LOCA w/Injection Failure	6.72E-07	3%
U1RDX	SGTR w/Failure to Terminate Leakage and Long Term Core Cooling Failure	4.70E-07	2%
Other		1.48E-06	6%
Total Freq:		2.3E-05	

Figure 3.7-5 System Importance

ST. LUCIE UNIT 1



ST. LUCIE UNIT 2



Human Reliability Analysis (HRA) Review

IPE for St. Lucie Nuclear Power Plant (Units 1 & 2)

Presented by

John Forester

September 11, 1996

Sandia National Laboratories

St. Lucie Nuclear Power Plant IPE HRA Overview

- Overall Impression Of This HRA Submittal
  - Average
- HRA Plant Vulnerabilities
  - None Identified
- Plant Improvements Identified Based On IPE Results
  - Implemented new AO to have operators fill CST from treated water storage tank when long-term operation (beyond 10 hours) of AFW is demanded.

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St. Lucie Nuclear Power Plant IPE HRA Overview (Continued)

- Licensee Participation

  - The HRA portion was performed entirely by utility personnel. The methodology was acquired from SAIC and used by FPL for Turkey Point and St. Lucie.

- Peer Review

  - PRA experts from ERIN Engineering, FRH, INC., NUS and Baltimore Gas & Electric (Level 1) contributed to a broad review of the entire PRA. Frank Hubbard from FRH and Niall Hunt from NUS were cited as reviewers knowledgeable in HRA.

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Pre-Initiator Human Events

- Types of Pre-initiators Considered

  - Pre-initiator Slips
    - Restoration after test, maintenance, or operational alignment
    - Miscalibration

- Identification Process

  - The systems analysis procedure directed analysts to include miscalibrations if, based on their understanding of the system design and operation, there were failures which could be significant contributors to the CDF

  - Maintenance, operating, and test procedures were reviewed. If the analyst determined that the maintained system, train, or component is not completely tested for its design function following maintenance, a failure to restore event was added to the fault trees. They were not modeled only if the components are realigned to correct configuration following a system actuation signal.

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Pre-Initiator Human Events (Continued)

- Identification Process (continued)

  - HRA focused on actions that might lead to failures of multiple trains of equipment, thus acting like common-cause failures."

    - Treated miscalibrations of all like instruments as common-cause.

    - Restorations modeled at train level - e.g., AFW pump 1A manual valve. Assumed maintenance etc. on separate trains was independent

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Pre-Initiator Human Events (Concluded)

- Screening Process for Pre-Initiator Human Failure Events (HFEs)

    - Pre-initiator HFEs (modeled in fault trees) were screened at a nominal value of 0.003 (per SAIC method used value from THERP). A beta factor of 0.1 is applied for multiple train events (miscalibrations only).

    - Licensee noted that the screening values were consistent with those used in another IPE found to be acceptable by the NRC

- Quantification Process (SAIC Method)

    - No additional quantification beyond the screening analysis.

    - Since only the screening analysis was performed, PSFs , self-recovery, and personnel redundancies/dependencies were not credited. Human common cause potential was evaluated.

    - Relative to some IPEs, pre-initiator HEPs would be somewhat pessimistic

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Post-Initiator Human Events

- Types of Events Considered
  - Time-independent post-initiator slips
  - Untimely responses (time-dependent): both in-control room and ex-control room
  - No explicit distinction between response and recovery was initially made in IPE. In response to RAI, response and recovery events were retrospectively categorized (quest. 7).

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Post-Initiator Human Events
(Continued)

- Identification Process

  - Response to RAI documents a reasonable process. Interviews with plant operations and training staff and reviews of EOPs, operating experience, other PRAs, and NRC PRA reviews were conducted. This info used in accident sequence and systems analysis task. Accident sequence event trees and top logic fault trees were developed. Human actions related to system operation were included in the top logic fault trees.

  - RAI indicated that events that fit into the recovery type were primarily identified during cutset review.

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Post-Initiator Human Events (Continued)

- Screening Process - None, human action events were set to 1.0 for initial quantification.

- Quantification - Two Techniques

  - Time-independent (similar to general application of SAIC time-independent model, but generally more thorough than most (see attachments 3, 4, 5, and 6 of response to RAI for examples) :

    - A basic human failure probability from NUREG/CR-1278 of 0.003
    - A dependency factor for other personnel, or a default of 1 (in the examples I examined, only took credit for one "recovery" and usually assumed high dependency (0.5). If moderate dependency was assumed used 0.14, if low, 0.05.
    - PSFs were multipliers based on NUREG/CR-1278 and the analyst's judgment. PSFs were only used to increase HEP (see Tables 3.4-2 and 3.4-3 of the submittal)

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Post-Initiator Human Events (Continued)

- Quantification -- Time-Independent Post-Initiator Slips (continued)
  - Problem:
    - Assumes time is not a factor and (at least in principle) that diagnosis failure is negligible (see question 13).
    - Approximately 50% of post-initiator events were treated as slips.
    - Analysis appeared thorough and conscientious (reasonable application of PSFs such as stress etc.) and values did not appear to be unreasonable.
    - Yet, approach parallels the IPEP method somewhat in that diagnosis is not explicitly quantified in these cases, <u>if it is clearly indicated by procedure</u>. Most other applications of the SAIC method that I have reviewed modeled (at most) only a couple post-initiator slips.
    - For example, once-through-cooling failure is treated as a slip.
    - Major difference between this and IPEP is reasonableness of HEPs, apparent level of analysis, and clear documentation.

Post-Initiator Human Events (Continued)

- Quantification (continued)
  - Time-Dependent - Standard SAIC TRC approach
    - In-control room technique makes use of the following parameters:
      - Net available time as the difference between total available time and other times as human factors considerations require
      - A type factor: 0.25 for verification actions; 0.5 for rule based actions; 1 for others
      - A success likelihood factor to reflect various performance shaping factors, or a default value of 0.5 which was done for this HRA
      - A burden factor: 1 with no burden; 2 otherwise
      - A model uncertainty factor, fixed at 1.68

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Post-Initiator Human Events (Continued)

- Quantification - Time-Dependent (continued)
  - Ex-control room technique makes use of the following parameters:
    - Net available time as the difference between total available time and other times as human factors considerations require
    - An estimate from operations personnel's judgment or walkdowns of the expected time to locate, access and manipulate the equipment
    - Additional time to reflect the potential delaying effects of specific types of ex-control room hazards
    - A model uncertainty factor, fixed at 1.68

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Post-Initiator Human Events
(Continued)

- Quantification "Notables"
  - Apparently used some actual estimates of median response times for in-control room actions (attachment 9).
  - Developed time lines for events and considered occurrence of cues (quest 15)
  - Where HEPs for same events might vary as a function of sequence, assumed worst case.
  - Some walkdowns occurred, but most response times estimated by operators
  - Considered sequence-specific dependencies (quest 17 and 18)
  - Modeled 2 non-proceduralized acts. for unit 1 (no data for unit 2). (quest 17)
  - For time-dependent modeled events, assume default SLI, thus assumed plant average in terms of PSFs. Similar to most IPEs which used this method.

Internal Flooding

- Some Operator Actions Apparently Considered, e.g., Terminate Flood.
  - No numerical credit was taken for special human actions discussed in submittal (level 1 question 14)
  - Seemingly, credit was taken for events already in model (e.g., trip caused by flooding) and values were not adjusted (no additional stress for flooding etc.)

Level 2 Analysis

- Recovery actions considered in level 1 were directly incorporated into PDS cutsets

- Apparently adjusted some LOSP recovery values

- Modeled operator actions to depressurize RCS - estimated HEP at 0.01 from in-control room TRC - proceduralized action

- Credited several other recovery actions - assigned "scoping" values of 0.5 to 0.05. (Quest. 9)

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St Lucie

St. Lucie 1& 2 Important Human Actions, Per F-V

1. Operator fails to secure RCPs following loss of seal cooling -- 3.1% of CDF

2. Common cause miscalibration of the RWT level transmitter -- 2.6% of CDF

3. Operator fails to restore power to Unit 1 from unit 2 --2.5% of CDF

4. Operators fail to do once through cooling for transient (Feed and Bleed) -- 2.3% of CDF

5. Operators fail to restore pump 1A after maintenance -- 1.6% of CDF

6. Operators fail to restore pump 1A after maintenance -- 1.5% of CDF

7. Operators fail to restore electrical equipment room fans following LOOP -- 1.5% CDF

8. Operators fail to do once through cooling for SGTR (Feed and Bleed) -- 0.7% of CDF

9. Operators fail to recover PCS following SIAS -- 0.7% of CDF

10. Operators fail to realign "AB" DC bus -- 0.6% of CDF

TABLE 3.4-1 PRE-INITIATOR HFE EVENTS					
UNIT	HFE BASIC EVENT	DESCRIPTION	FAILURE MODE	PROBABILITY	ERROR FACTOR
1	AHFL109108	AFW PUMP 1A MANUAL VALVE V09108 MISPOSITIONED	SLIP	3.00E-03	10
1	AHFL109124	AFW PUMP 1B MANUAL VALVE V09124 MISPOSITIONED	SLIP	3.00E-03	10
1	AHFL109140	AFW PUMP 1C MANUAL VALVE V09140 MISPOSITIONED	SLIP	3.00E-03	10
1	BHFL1LVL	COMMON CAUSE FAILURE OF SIT'S DUE TO MISCALIBRATION OF SIT LEVEL SENSORS	SLIP	3.00E-04	10
1	BHFL1PRS	COMMON CAUSE FAILURE OF SIT'S DUE TO MISCALIBRATION OF SIT PRESSURE SENSORS	SLIP	3.00E-04	10
1	DHFL1HVS1A	OPERATOR FAILS TO RESTORE HVS 1A FOLLOWING MAINTENANCE	SLIP	3.00E-03	10
1	DHFL1HVS1B	OPERATOR FAILS TO RESTORE HVS 1B FOLLOWING MAINTENANCE	SLIP	3.00E-03	10
1	DHFL1HVS1C	OPERATOR FAILS TO RESTORE HVS 1C FOLLOWING MAINTENANCE	SLIP	3.00E-03	10
1	DHFL1HVS1D	OPERATOR FAILS TO RESTORE HVS 1D FOLLOWING MAINTENANCE	SLIP	3.00E-03	10
1	EHFL1EDG1A	OPERATOR FAILS TO PROPERLY ALIGN FOLLOWING MAINTENANCE	SLIP	3.00E-03	10
1	EHFL1EDG1B	OPERATOR FAILS TO PROPERLY ALIGN FOLLOWING MAINTENANCE	SLIP	3.00E-03	10
1	GHFL1PUMPA	OPERATOR FAILS TO RESTORE PUMP 1A FOLLOWING MAINTENANCE	SLIP	3.00E-03	10
1	GHFL1PUMPB	OPERATOR FAILS TO RESTORE PUMP 1B FOLLOWING MAINTENANCE	SLIP	3.00E-03	10
1	HHFL1STBYA	OPERATOR FAILS TO PUT CTMT AIR COMPRESSOR 1A IN STANDBY	SLIP	3.00E-03	10
1	HHFL1STBYB	OPERATOR FAILS TO PUT CTMT AIR COMPRESSOR 1B IN STANDBY	SLIP	3.00E-03	10
1	HHFL1STBYC	OPERATOR FAILS TO PUT AIR COMPRESSOR 1C IN STANDBY	SLIP	3.00E-03	10
1	HHFL1STBYD	OPERATOR FAILS TO PUT AIR COMPRESSOR 1D IN STANDBY	SLIP	3.00E-03	10
1	JHFL1PUMPA	OPERATOR FAILS TO RESTORE PUMP 1A FOLLOWING MAINTENANCE	SLIP	3.00E-03	10
1	JHFL1PUMPB	OPERATOR FAILS TO RESTORE PUMP 1B FOLLOWING MAINTENANCE	SLIP	3.00E-03	10
1	JHFL1SDA	OPERATOR FAILS TO RESTORE SDC HX 1A FOLLOWING MAINTENANCE	SLIP	3.00E-03	10
1	JHFL1SDB	OPERATOR FAILS TO RESTORE SDC HX 1B FOLLOWING MAINTENANCE	SLIP	3.00E-03	10
1	LHFL1PUMPA	OPERATOR FAILS TO PROPERLY RESTORE CS PUMP A FOLLOWING MAINTENANCE	SLIP	3.00E-03	10
1	LHFL1PUMPB	OPERATOR FAILS TO PROPERLY RESTORE CS PUMP B FOLLOWING MAINTENANCE	SLIP	3.00E-03	10
1	MHFL1V2154	VALVE V2154 LEFT MISPOSITIONED FOLLOWING MAINTENANCE	SLIP	3.00E-03	10

TABLE 3.4-1
PRE-INITIATOR HFE EVENTS

UNIT	HFE BASIC EVENT	DESCRIPTION	FAILURE MODE	PROBABILITY	ERROR FACTOR
1	MHFL1V2155	VALVE V2155 LEFT MISPOSITIONED FOLLOWING MAINTENANCE	SLIP	3.00E-03	10
1	NHFL1PPCCF	COMMON CAUSE MISCALIBRATION OF PRESSURIZER PRESSURE TRANSMITTERS	SLIP	3.00E-04	10
1	NHFL1RWLCF	COMMON CAUSE MISCALIBRATION OF THE RWT LEVEL TRANSMITTERS	SLIP	3.00E-04	10
1	NHFL1SGPCF	COMMON CAUSE MISCALIBRATION OF THE STEAM GENERATOR PRESSURE TRANSMITTERS	SLIP	3.00E-04	10
1	NHFL1CPCCF	COMMON CAUSE MISCALIBRATION OF THE CONTAINMENT PRESSURE TRANSMITTERS	SLIP	3.00E-04	10
1	NHFL1CRMCF	COMMON CAUSE MISCALIBRATION OF THE CONTAINMENT RADIATION MONITORS	SLIP	3.00E-04	10
2	AHFL209108	AFW PUMP 2A MANUAL VALVE V09108 MISPOSITIONED	SLIP	3.00E-03	10
2	AHFL209124	AFW PUMP 2B MANUAL VALVE V09124 MISPOSITIONED	SLIP	3.00E-03	10
2	AHFL209140	AFW PUMP 2C MANUAL VALVE V09140 MISPOSITIONED	SLIP	3.00E-03	10
2	BHFL2LVL	COMMON CAUSE FAILURE OF SIT'S DUE TO MISCALIBRATION OF SIT LEVEL SENSORS	SLIP	3.00E-04	10
2	BHFL2PRS	COMMON CAUSE FAILURE OF SIT'S DUE TO MISCALIBRATION OF SIT PRESSURE SENSORS	SLIP	3.00E-04	10
2	DHFL2HVS1A	OPERATOR FAILS TO RESTORE HVS 1A FOLLOWING MAINTENANCE	SLIP	3.00E-03	10
2	DHFL2HVS1B	OPERATOR FAILS TO RESTORE HVS 1B FOLLOWING MAINTENANCE	SLIP	3.00E-03	10
2	DHFL2HVS1C	OPERATOR FAILS TO RESTORE HVS 1C FOLLOWING MAINTENANCE	SLIP	3.00E-03	10
2	DHFL2HVS1D	OPERATOR FAILS TO RESTORE HVS 1D FOLLOWING MAINTENANCE	SLIP	3.00E-03	10
2	EHFL2EDG2A	OPERATOR FAILS TO PROPERLY ALIGN FOLLOWING MAINTENANCE	SLIP	3.00E-03	10
2	EHFL2EDG2B	OPERATOR FAILS TO PROPERLY ALIGN FOLLOWING MAINTENANCE	SLIP	3.00E-03	10
2	GHFL2PUMPA	OPERATOR FAILS TO RESTORE PUMP 1A FOLLOWING MAINTENANCE	SLIP	3.00E-03	10
2	GHFL2PUMPB	OPERATOR FAILS TO RESTORE PUMP 1B FOLLOWING MAINTENANCE	SLIP	3.00E-03	10
2	HHFL2STBYC	OPERATOR FAILS TO PUT AIR COMPRESSOR 1C IN STANDBY	SLIP	3.00E-03	10
2	HHFL2STBYD	OPERATOR FAILS TO PUT AIR COMPRESSOR 1D IN STANDBY	SLIP	3.00E-03	10
2	JHFL2PUMPA	OPERATOR FAILS TO RESTORE PUMP 2A FOLLOWING MAINTENANCE	SLIP	3.00E-03	10
2	JHFL2PUMPB	OPERATOR FAILS TO RESTORE PUMP 2B FOLLOWING MAINTENANCE	SLIP	3.00E-03	10

TABLE 3.4-1 PRE-INITIATOR HFE EVENTS					
UNIT	HFE BASIC EVENT	DESCRIPTION	FAILURE MODE	PROBABILITY	ERROR FACTOR
2	JHFL2SDCA	OPERATOR FAILS TO RESTORE SDC HX 2A FOLLOWING MAINTENANCE	SLIP	3.00E-03	10
2	JHFL2SDCB	OPERATOR FAILS TO RESTORE SDC HX 2B FOLLOWING MAINTENANCE	SLIP	3.00E-03	10
2	LHFL2PUMPA	OPERATOR FAILS TO PROPERLY RESTORE CS PUMP A FOLLOWING MAINTENANCE	SLIP	3.00E-03	10
2	LHFL2PUMPB	OPERATOR FAILS TO PROPERLY RESTORE CS PUMP B FOLLOWING MAINTENANCE	SLIP	3.00E-03	10
2	MHFL2V2154	VALVE V2154 LEFT MISPOSITIONED FOLLOWING MAINTENANCE	SLIP	3.00E-03	10
2	MHFL2V2155	VALVE V2155 LEFT MISPOSITIONED FOLLOWING MAINTENANCE	SLIP	3.00E-03	10
2	NHFL2PPCCF	COMMON CAUSE MISCALIBRATION OF PRESSURIZER PRESSURE TRANSMITTERS	SLIP	3.00E-04	10
2	NHFL2RWLCF	COMMON CAUSE MISCALIBRATION OF THE RWT LEVEL TRANSMITTERS	SLIP	3.00E-04	10
2	NHFL2SPCCF	COMMON CAUSE MISCALIBRATION OF THE STEAM GENERATOR PRESSURE TRANSMITTERS	SLIP	3.00E-04	10
2	NHFL2CPCCF	COMMON CAUSE MISCALIBRATION OF THE CONTAINMENT PRESSURE TRANSMITTERS	SLIP	3.00E-04	10
2	NHFL2CRMCF	COMMON CAUSE MISCALIBRATION OF THE CONTAINMENT RADIATION MONITORS	SLIP	3.00E-04	10

TABLE 3.4-2 ST. LUCIE UNIT 1 POST-INITIATOR HFE

UNIT	RECOVERY EVENT	DESCRIPTION	PROBABILITY	ERROR FACTOR	LOCATION OF ACTION	FAILURE MODE	PSF	TYPE OF BEHAVIOR	BURDEN	RESPONSE TIME (MIN)	AVAILABLE TIME (MIN)	TIMING SOURCE	EVALUATION TYPE
1	RBAFWCMP	OPERATOR FAILS TO ACTUATE AFW COMPONENT(S)	3.00E-03	10	IN CR	SLIP	1	NA	NA	NA	NA	NA	HU
1	RBAFWXVLVS	OPERATOR FAILS TO MANUALLY OPEN AFW X-TIE VLVS AND SG FLOW VLVS	3.64E-02	4.4	EX CR	INADEQUATE RESPONSE	NA	NA	NA	10	55	1	HU
1	RBDCAB	OPERATOR FAILS TO REALIGN POWER SUPPLY TO 125VDC BUS AB	3.57E-03	3	IN CR	MISTAKE	NA	RESPONSE	NO	4	30	1,3	HU
1	RBDGFO	OPERATOR FAILS TO OPEN DG FO FP.L VALVE BYPASS	3.69E-02	4.4	EX CR	INADEQUATE RESPONSE	NA	NA	NA	10	55	1	HU
1	RBPCSIAS	OPERATOR FAILS TO RESTORE PCS FOLLOWING ESFAS ACTUATION	3.5E-01	10	IN CR/EX CR	SLIP/ INADEQUATE RESPONSE	1	NA	NA	EX CR - 30	EX CR - 43	1	HU
1	RBPCSMFW	OPERATOR FAILS TO RECOVER MAIN FEED WATER	1.21E-03	3.2	IN CR	MISTAKE	NA	RESPONSE	NO	5	55	1,3	HU
1	RAFW1CST	OPERATOR FAILS TO SWITCH AFW TO UNIT 2 CST	3.15E-04	4.4	EX CR	INADEQUATE RESPONSE	NA	NA	NA	20	520	1,2	HU
1	RAFW1SGTR	OPERATOR FAILS TO REALIGN AFW AND ISOLATE THE FAULTED SG FOLLOWING SGTR	3.00E-03	10	IN CR	SLIP	1	NA	NA	NA	NA	NA	HU
1	RCSS1CSAS	OPERATOR FAILS TO ACTUATE CSS COMPONENTS	6.00E-03	10	IN CR	SLIP	2	NA	NA	NA	NA	NA	HU
1	RCST1TWST	OPERATOR FAILS TO PROVIDE LONG TERM MAKEUP TO CST VIA TWST	1.55E-05	4.4	EX CR	INADEQUATE RESPONSE	NA	NA	NA	10	530	1,3	HU
1	RCVC1RWT	OPERATOR FAILS TO SWITCH CHARGING PUMP SUCTION TO RWT	3.00E-03	10	IN CR	SLIP	1	NA	NA	NA	NA	NA	HU
1	REPS1XTIS	OPERATOR FAILS TO RESTORE POWER TO UNIT 1 FROM UNIT 2	1.88E-02	6.4	IN CR	MISTAKE	NA	RULE	YES	10	60	1	HU
1	RHPS1RAS	OPERATOR FAILS TO MANUALLY ACTUATE COMPONENT FOLLOWING RAS	1.50E-03	10	IN CR	SLIP	1	NA	NA	NA	NA	NA	HU
1	RHPS1SIAS	OPERATOR FAILS TO MANUALLY ACTUATE COMPONENT FOLLOWING SIAS	3.00E-03	10	IN CR	SLIP	1	NA	NA	NA	NA	NA	HU
1	RHVA1ELEQ	OPERATOR FAILS TO RESTORE ELECTRICAL EQUIPMENT ROOM FANS FOLLOWING LOSS OF PWR	5.59E-03	10	IN CR/EX CR	MISTAKE/ INADEQUATE RESPONSE	1	RULE	NO	IN CR - 30/ EX CR - 5	IN CR - 120/ EX CR - 90	1,2,4	HU
1	RIA1AB	OPERATOR FAILS TO ALIGN 1A OR 1B IA COMPRESSOR	~~[illegible]~~ 1.23E-2	10	IN CR/EX CR	MISTAKE/ INADEQUATE RESPONSE	1	RULE	NA	IN CR - 120/ EX CR - 45	1080	1,2	HU
1	RLPS1HTLEG	OPERATOR FAILS TO INITIATE HOT LEG RECIRC	7.50E-04	10	IN CR	SLIP	1	NA	NA	NA	NA	NA	HU
1	RPCS1XFSD	OPERATOR FAILS TO RESTORE PCS FOLLOWING EXCESSIVE FEEDING OF SG'S	1.67E-02	3.2	IN CR	MISTAKE	NA	RESPONSE	NO	11	60	1	HU

TABLE 3.4-2 (Cont'd) ST. LUCIE UNIT 1 POST-INITIATOR HFE

UNIT	RECOVERY EVENT	DESCRIPTION	PROBABILITY	ERROR FACTOR	LOCATION OF ACTION	FAILURE MODE	PSF	TYPE OF BEHAVIOR	BURDEN	RESPONSE TIME (MIN)	AVAILABLE TIME (MIN)	TIMING SOURCE	EVALUATION TYPE
1	RPPC1-PORV	CONTROL ROOM OPERATOR FAILS TO ISOLATE PORV PATH	3.00E-03	10	IN CR	SLIP	2	NA	NA	NA	NA	NA	HU
1	RPPC1BLPWR	OPERATOR FAILS TO RESTORE POWER TO PORV BLOCK VALVE	2.04E-01	4.4	EX CR	INADEQUATE RESPONSE	NA	NA	NA	5	11	1,1	HU
1	RTOP1RLTC	OPERATOR FAILS TO IMPLEMENT SHUTDOWN COOLING (SGTR)	7.50E-04	10	IN CR	SLIP	1	NA	NA	NA	NA	NA	HU
1	RTOP1ROTC	OPERATOR FAILS TO INITIATE ONCE-THROUGH COOLING FOR SGTR	7.50E-03	10	IN CR	SLIP	5	NA	NA	NA	NA	NA	HU
1	RTOP1SIOTC	OPERATOR FAILS TO INITIATE ONCE-THROUGH COOLING FOR SI LOCA	7.50E-03	10	IN CR	SLIP	5	NA	NA	NA	NA	NA	HU
1	RTOP1SIRCP	OPERATOR FAILS TO SECURE RCP'S FOLLOWING LOSS OF SEAL COOLING	3.00E-04	10	IN CR	SLIP	2	NA	NA	NA	NA	NA	HU
1	RTOP1TERM	OPERATOR FAILS TO TERMINATE LEAKAGE ON FAULTED SG	7.50E-04	10	IN CR	SLIP	1	NA	NA	NA	NA	NA	HU
1	RTOP1TLTC	OPERATOR FAILS TO IMPLEMENT SDC (TRANSIENT)	4.38E-03	3.2	IN CR	MISTAKE	NA	RESPONSE	NO	180	1440	1	HU
1	RTOP1TOTC	OPERATOR FAILS TO INITIATE ONCE THROUGH COOLING FOR TRANSIENT	7.50E-03	10	IN CR	SLIP	5	NA	NA	NA	NA	NA	HU
1	RTOP1WBOR	OPERATOR FAILS TO BORATE DURING ATWS	6.00E-03	10	IN CR	SLIP	4	NA	NA	NA	NA	NA	HU
1	RTOPWLTC	OPERATOR FAILS TO IMPLEMENT SDC (ATWS)	4.38E-03	3.2	IN CR	MISTAKE	NA	RESPONSE	NO	180	1440	1	HU

TIMING SOURCE KEY:
1 - OPERATOR ESTIMATE
2 - WALKDOWN
3 - MAPP
4 - PROCEDURES

EVALUATION TYPE KEY:
HU - HUMAN

TABLE 3.4-3 ST. LUCIE UNIT 2 POST-INITIATOR HFE

UNIT	RECOVERY EVENT	DESCRIPTION	PROBABILITY	ERROR FACTOR	LOCATION OF ACTION	FAILURE MODE	PSF	TYPE OF BEHAVIOR	BURDEN	RESPONSE TIME (MIN)	AVAILABLE TIME (MIN)	TIMING SOURCE	EVALUATION TYPE
2	R8AFWCMP	OPERATOR FAILS TO ACTUATE AFW COMPONENTS	3.00E-03	10	IN CR	SLIP	1	NA	NA	NA	NA	NA	HU
2	R8AFXVLVS	OPERATOR FAILS TO UTILIZE AFW X-CONNECT VLVS	3.69E-02	4.4	EX CR	INADEQUATE RESPONSE	NA	NA	NA	10	35	1	HU
2	R82PCSMFW	OPERATOR FAILS TO RECOVER MAIN FEED WATER	1.42E-03	3.2	IN CR	MISTAKE	NA	RESPONSE	NO	5	33	1,3	HU
2	R8DCAB	OPERATOR FAILS TO REALIGN POWER SUPPLY TO 125VDC BUS AB	5.57E-03	3.2	IN CR	MISTAKE	NA	RESPONSE	NO	4	30	1,3	HU
2	R8DGFO	OPERATOR FAILS TO RECOVER EDG BY OPENING DG FILL VALVE BYPASS	3.69E-02	4.4	EX CR	INADEQUATE RESPONSE	NA	NA	NA	10	35	1	HU
2	R8PCSIAS	OPERATOR FAILS TO RESTORE PCS FOLLOWING ESFAS ACTUATION	1.50E-03	10	IN CR	SLIP	1	NA	NA	NA	NA	NA	HU
2	RAFW2SOTR	OPERATOR FAILS TO REALIGN AFW AND ISOLATE THE FAULTED SG FOLLOWING SGTR	3.00E-03	10	IN CR	SLIP	1	NA	NA	NA	NA	NA	HU
2	RCVC2RWT	OPERATOR FAILS TO SWITCHCHARGING PUMP SUCTION TO RWT	3.03E-03	10	IN CR	SLIP	1	NA	NA	NA	NA	NA	HU
2	REPS2XT1B	OPERATOR FAILS TO RESTORE POWER TO UNIT 2 FROM UNIT 1	1.89E-02	6.4	IN CR	MISTAKE	NA	RULE	YES	10	60	1	HU
2	RHPS2HTLEG	OPERATOR FAILS TO INITIATE HOT LEG RECIRC	5.31E-03	10	IN CR/EX CR	SLIP/ INADEQUATE RESPONSE	1	NA	NA	EX CR 30	EX CR 360	1,4	HU
2	RHPS2RAS	OPERATOR FAILS TO MANUALLY ACTUATE COMPONENT FOLLOWING RAS	1.50E-03	10	IN CR	SLIP	2	NA	NA	NA	NA	NA	HU
2	RHPS2SIAS	OPERATOR FAILS TO MANUALLY ACTUATE COMPONENT FOLLOWING SIAS	3.0E-03	10	IN CR	SLIP	1	NA	NA	NA	NA	NA	HU
2	RHVA2ELEQ	OPERATOR FAILS TO RESTORE ELECTRICAL EQUIPMENT ROOM FANS FOLLOWING LOSS OF PWR	4.20E-02	3.2	IN CR	MISTAKE	1	RESPONSE	NO	IN CR 30	IN CR 120	1,2	HU
2	RPCS2XFSD	OPERATOR FAILS TO RESTORE PCS FOLLOWING EXCESSIVE FEEDING OF SG'S	1.67E-02	3.2	IN CR	MISTAKE	NA	RESPONSE	NO	11	60	1	HU
2	RPFC2-PORV	CONTROL ROOM OPERATOR FAILS TO ISOLATE PORV PATH	3.90E-03	10	IN CR	SLIP	2	NA	NA	NA	NA	NA	HU
2	RPFC2BLPWR	OPERATOR FAILS TO RESTORE POWER TO PORV BLOCK VALVE	2.04E-01	4.4	EX CR	INADEQUATE RESPONSE	NA	NA	NA	5	11	1,2	HU
2	RTOP2RLTC	OPERATOR FAILS TO IMPLEMENT SHUTDOWN COOLING (SGTR)	7.50E-04	10	IN CR	SLIP	1	NA	NA	NA	NA	NA	HU
2	RTOP2ROTC	OPERATOR FAILS TO INITIATE ONCE-THROUGH COOLING FOR SGTR	7.50E-03	10	IN CR	SLIP	1	NA	NA	NA	NA	NA	HU

TABLE 3.4-3 (Cont'd) ST. LUCIE UNIT 2 POST-INITIATOR HFE

UNIT	RECOVERY EVENT	DESCRIPTION	PROBABILITY	ERROR FACTOR	LOCATION OF ACTION	FAILURE MODE	PSF	TYPE OF BEHAVIOR	BURDEN	RESPONSE TIME (MIN)	AVAILABLE TIME (MIN)	TIMING SOURCE	EVALUATION TYPE
2	RTOP2SIOTC	OPERATOR FAILS TO INITIATE ONCE-THROUGH-COOLING FOR SI LOCA	7.50E-03	10	IN CR	SLIP	1	NA	NA	NA	NA	NA	HU
2	RTOP2SIRCP	OPERATOR FAILS TO SECURE RCP'S FOLLOWING LOSS OF SEAL COOLING	3.00E-04	10	IN CR	SLIP	2	NA	NA	NA	NA	NA	HU
2	RTOP2TOTC	OPERATOR FAILS TO INITIATE ONCE THROUGH COOLING FOR TRANSIENT	7.50E-03	10	IN CR	SLIP	5	NA	NA	NA	NA	NA	HU
2	RTOP2WBOR	OPERATOR FAILS TO BORATE DURING ATWS	6.00E-03	10	IN CR	SLIP	4	NA	NA	NA	NA	NA	HU

TIMING SOURCE KEY:

1 - OPERATOR ESTIMATE

2 - WALKDOWN

3 - MAPP

4 - PROCEDURES

EVALUATION TYPE KEY:

HU - HUMAN