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January 27, 1995

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ULNRC-3133

Gentlemen:

CALLAWAY PLANT
DOCKET NUMBER 50-483
EXEMPTION TO 10 CFR 50, APPENDIX J AND REVISION TO TECHNICAL SPECIFICATION 4.6.1.2.a - CONTAINMENT SYSTEMS

Reference: ULNRC-3112 dated December 9, 1994

The referenced letter transmitted a request for a 10 CFR 50.12 exemption to 10 CFR 50, Appendix J and an application for amendment to Facility Operating License Number NPF-30. The exemption and related amendment would defer the next scheduled containment integrated leak rate test from Refuel 7 to Refuel 8.

The attachment to this letter provides additional plant specific information in support of the subject exemption request as follows:

Executive Summary

Appendix I: Callaway Containment Structure Description

Appendix II: Containment Building Overpressure Capability

Appendix III: Probabilistic Risk Evaluation of ILRT Deferral

Appendix IV: Assessment of ILRT Benefits and Risks

This material is submitted in support of our contention that deferral of the ILRT to Refuel 8 will not impact public health and safety.

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It should be noted that Union Electric will continue to perform local leak rate testing in accordance with the requirements of 10 CFR 50 Appendix J. We will also perform the general containment inspection during Refuel 7 which commences in March, 1995.

If you have any questions concerning this information, please contact us.

Very truly yours,

Donald F. Schnell

Donald F. Schnell

DFS/bjp

Attachment

STATE OF MISSOURI)
) S S
CITY OF ST. LOUIS)

Donald F. Schnell, of lawful age, being first duly sworn upon oath says that he is Senior Vice President-Nuclear and an officer of Union Electric Company; that he has read the foregoing document and knows the content thereof; that he has executed the same for and on behalf of said company with full power and authority to do so; and that the facts therein stated are true and correct to the best of his knowledge, information and belief.

By Donald F. Schnell
Donald F. Schnell
Senior Vice President
Nuclear

SUBSCRIBED and sworn to before me this 27th day of January, 1995.

Barbara J. Pfaff
BARBARA J. PFAFF
NOTARY PUBLIC — STATE OF MISSOURI
MY COMMISSION EXPIRES APRIL 22, 1997
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ADDITIONAL INFORMATION IN SUPPORT OF
10 CFR 50.12 EXEMPTION TO 10 CFR 50, APPENDIX J
AND REVISION TO TECHNICAL SPECIFICATION 4.6.1.2.a

EXECUTIVE SUMMARY

The information transmitted in this attachment supplements the 10 CFR 50.12 exemption to 10 CFR 50, Appendix J and revision to Technical Specification 4.6.1.2.a as transmitted by Union Electric letter dated December 9, 1994 (ULNRC-3112). Included in this attachment are Appendices I through IV AS follows:

Appendix I describes the Callaway containment design. The information contained in this appendix confirms that the Callaway containment building is an unusually robust, large, dry containment with no weak links with respect to challenges presented by severe accidents.

Appendix II summarizes the ultimate strength capability of the Callaway containment structure when subjected to severe temperatures and pressures beyond design values. Failure probabilities are estimated with 95%, 50%, and 5% confidence levels for internal temperatures up to 400°F using methodologies previously accepted by the NRC. Because of the high assessed strength of the containment, early overpressure failure is very unlikely, and long term overpressure failure is not predicted to occur until at least 48 hours in most cases.

Appendix III consists of excerpts from the Callaway IPE which determine the source term probabilistic risk impact associated with deferring the ILRT to Refuel 8. This appendix concludes that there is negligible impact on offsite dose due to ILRT frequency, given the insensitivity of risk to containment leak rate. Therefore, the deferral of the ILRT until Refuel 8 will result in no increased risk to the general public.

Appendix IV shows that the additional data obtained during an ILRT is costly and of little value when compared to the data obtained from Type B and C programs (LLRTs). The LLRTs provide a more accurate estimate of overall containment leakage than does the ILRT. This appendix also discusses the potential for human error during performance of an ILRT, which far exceeds that associated with any other plant evolution or test. Industry data is presented to illustrate this negative aspect of an ILRT and its potential consequences. We believe the risk of mispositioning or misaligning a valve or system is greater than the benefit gained by performing an ILRT.

This information (in conjuction with the December 9, 1994 letter), supports our contention that the Callaway containment is well built, robust, essentially leak tight and has considerable ultimate strength margin. Since the majority of ILRT failures are directly attributed to leakage through containment penetrations, which are identified and corrected by local leak rate testing, we conclude that deferring the ILRT to Refuel 8 in September, 1996 will not involve any unreviewed safety questions or increased risk to the public.

LIST OF ACRONYMS

CD	Core Damage Sequence
CET	Containment Event Tree
CFR	Code of Federal Regulations
DCH	Direct Containment Heating
DF	Decontamination Factor
ESFAS	Engineered Safety Features Actuation System
ET	Event Tree
IDCOR	Industry Degraded Core Rulemaking Program
ILRT	Integrated Leak Rate Test
IPE	Individual Plant Examination
ISL	Interfacing Systems LOCA
LER	Licensee Event Report
LLRT	Local Leak Rate Test
LOCA	Loss of Coolant Accident
MAAP	Modular Accident Analysis Program
MSIV	Main Steam Isolation Valve
PDS	Plant Damage Status
PORV	Power Operated Relief Valve
RHR	Residual Heat Removal
RPS	Reactor Protection System
SGTR	Steam Generator Tube Rupture
STC	Source Term Category

APPENDIX I - CALLAWAY CONTAINMENT STRUCTURE DESCRIPTION

The Callaway containment building is a prestressed, post-tensioned concrete structure which houses the reactor vessel, the reactor coolant system and supporting primary systems. The structure consists of a reinforced concrete, right circular cylinder with a hemispherical dome, and an integrally constructed, reinforced concrete base slab. The containment building is designed to control the release of airborne radioactivity following postulated design basis accidents and to provide shielding for the reactor core and coolant system. The interior of the containment structure is maintained at near pressure during normal power operation.

The structure is supported on stabilized backfill placed over a conglomeration of clay and chert having high load bearing properties. The post-tensioning of the wall and dome is accomplished through vertical tendons extending the full height of the wall and over the dome, and horizontal circumferential tendons around the wall and a portion of the dome. The base slab is below plant grade with the containment walls and operating floor above grade. The inside of the structure is lined with steel plate to form an essentially leak tight barrier. The containment structure is designed for a leakage rate not to exceed two-tenths weight percent of the free volume per day for the first 24 hours of a postulated accident and one-tenth weight percent per day thereafter at the design pressure of 60 psig. The interior free volume is approximately 2.5 million cubic feet.

The containment base slab is a 10 ft. thick reinforced concrete cylinder. Below the base is a continuous peripheral tendon access gallery for the installation and inspection of the vertical post-tensioning system. Additional reinforcing is provided at discontinuities in the structure and at major penetrations in the shell.

The post-tensioning system consists of unbonded tendons, each comprised of 170 one-fourth inch diameter high strength wires. Each tendon has an ultimate strength of approximately 1,000 tons. There are 86 inverted U-shaped tendons that extend through the full height of the cylindrical wall, continue over the dome, and are anchored at the bottom of the base mat. Inverted U-shaped tendons form a two-way pattern over the top of the dome. There are also 135 circumferential (horizontal) tendons spaced along the height of the shell, and an additional 30 horizontal tendons in the lower portion of the dome up to a height of an approximate 45-degree vertical angle from the springline. Each of the horizontal tendons extends 240 degrees around the shell and is anchored at the buttresses. This results in an effective prestress which is slightly greater than an equivalent uniform negative pressure of 72 psig.

Principal nominal dimensions of the reactor building are as follows:

Internal diameter	140 ft.
Interior height	205 ft.
Height to spring line	135 ft.
Vertical wall thickness	4 ft.
Dome thickness	3 ft.
Base slab thickness	10 ft.
Base slab diameter	154 ft.
Liner plate thickness	0.25 in.
Internal free volume	2.5×10^6 cubic ft.

Liner Plate System

A carbon steel liner plate covers the entire inside surface of the reactor building (excluding penetrations). The 1/4 inch thick liner is thickened locally from 1/2 to 2 inches around penetrations, large brackets, and major attachments. The liner plate, including the thickened plate, is anchored to the concrete structure.

Attachments to the liner plate which transfer loads through the liner plate to the base slab include equipment support anchors and reinforcing steel for support of the internal structures. Major structural attachments to the walls which penetrate the liner plate include polar crane brackets, floor beam brackets, and pipe support brackets. Major structural attachments to the dome include various pipe support brackets.

The number of liner penetrations is limited in order to minimize the potential for leakage. All penetrations are leak-tight assemblies welded to the steel liner. The penetrations include:

- a personnel air lock and an auxiliary personnel hatch
- an equipment hatch
- 81 piping penetrations
- 55 electrical penetrations
- the fuel transfer tube penetration
- 2 purge line penetrations

Mini-purge penetrations are designed to fail-closed. The analysis performed to support the Callaway IPE shows that this feature makes the isolation system extremely effective.

Conclusion

The Callaway containment building is a large, dry containment structure having no weak links. The fail-closed design of major piping penetrations combined with the building structural design results in a highly reliable, low leakage system having a low probability of failure to isolate (see Reference 1). Based on the design and past ILRT results, Union Electric is confident

that the ILRT currently scheduled for Refuel 7 can be deferred until Refuel 8 without impacting either Appendix J or Technical Specification leakage limits.

<u>References</u>

1. EPRI TR-104285, Risk Impact Assessment of Revised Containment Leak Rate Testing Intervals, August 1994.

APPENDIX II - CONTAINMENT BUILDING OVERPRESSURE CAPABILITY

The Callaway Individual Plant Examination Level II analysis considers the possibility of the containment building failing under various severe accident scenarios. The IPE characterized containment failure using four parameters: likelihood of failure as a function of containment pressure, failure size, location of failure, and timing. Likelihood of failure at a given pressure is the primary parameter of interest in the study. Failure size is important because the larger the opening, the faster the release of radionuclides following an accident. The location of the failure is important because the retention of radioactive materials can be dependent on this parameter. For a similar reason, timing is also important. The longer the materials can be retained inside the containment before escaping, the larger the reduction in source term to the environment since the radionuclides are removed from the containment atmosphere by natural processes and spray operation.

The ultimate containment failure pressure has been explicitly analyzed for the Callaway containment by Bechtel Power Corporation. Since the probability of global detonations was judged to be quite small, only static loads were treated. The containment strength analysis identified the median ultimate pressure capacity to be 134.9 psig (at 50% confidence level) and the lower bound to be 126.9 psig (at 95% reliability with 50% confidence). It is also emphasized that these failure pressures represent the threshold of yielding of the liner, horizontal pre-stressed tendons, and horizontal reinforcement, at which point the structure begins losing a large amount of its stiffness. No breaching or leakage occurs at this point. The following summarizes the fragility curves developed by this analysis:

Pressure (psia)	Failure Probability with 50% Confidence	Failure Probability with 95% Confidence
120	0.0	0.1
130	0.005	0.27
140	0.03	0.65
150	0.55	0.99
160	0.98	1.0

The above values were determined at 400°F. As an upper bound, a sensitivity analysis evaluated the impact of temperatures up to 800°F. The containment strength and associated failure probabilities were not significantly affected by increased temperatures.

Conclusion

The containment ultimate strength analysis determined that the Callaway containment can withstand pressures of roughly twice the design pressure. We therefore conclude that the Callaway containment building structure is robust with respect to challenges presented by severe accidents. Deferring the ILRT until Refuel 8 will have no impact on this assessment.

APPENDIX III - PROBABILISTIC RISK EVALUATION OF ILRT DEFERRAL

To evaluate the probabilistic risk impact of ILRT deferral we utilized the existing Callaway IPE Level II models. Because the Callaway IPE is a Level II PRA it is not possible to directly assess offsite consequences. Therefore, this evaluation focuses primarily on changes to source terms and their frequency. The evaluation shows that while the frequency associated with certain source terms will change, these source terms are small and do not contribute to the overall hazard to the public. The evaluation is carried out in two parts. The first part focuses on the change in probability for any source term due to deferring the ILRT. The second part evaluates the source terms impacted by deferral and judges this impact on the public.

Simplified Risk Model

The Callaway IPE Containment Event Tree (CET) shown in Figure 2 provides the starting point to evaluate the effect of deferring the ILRT on risk of offsite release and the source term. The Callaway IPE CET does not include certain failure modes required to evaluate deferral of ILRT testing. These failure modes involve failure of containment isolation components to fully seal. Our IPE does not consider these failure modes for two reasons: first, best estimate containment leakage was modeled which is a small leakage source term; second, perturbation of this small source term was unnecessary since other Level III PRAs have shown that leakage results in small radiological releases which have little impact on the public when compared to other severe accident source terms.

The impact of ILRT deferral can be evaluated without resorting to complex CET models. Therefore, a simplified event tree was developed to distinguish those accident sequences affected by the status of the containment isolation system versus those that are a direct function of severe accident phenomena. The simplified event tree permits a smaller number of CET scenarios to be evaluated to determine the baseline risk for Callaway as well as subsequent risk effect due to extending ILRT test intervals. Risk in this context is defined as both the frequency of a release and its source term.

Risk Characteristics

Only a few risk characteristics are influenced by Appendix J leak rate testing requirements. Key parameters include core damage accident sequences that are coupled with containment penetrations or isolation system failures, and leakage from an intact containment. The risks of primary concern are:

a) Accident sequences that do not lead to containment failure with an initially isolated containment (the containment remains intact in the long term);

b) Accidents in which the containment fails to isolate;
c) Accident groups involving containment failure induced by severe accident phenomena; and,
d) Accidents in which the containment is bypassed.

Accident sequences that do not lead to containment failure (group a, above) are influenced by the assumed leakage rate of the containment, given successful isolation of the penetrations. Accident sequences with an initially unisolated containment (i.e., containment fails to isolate (group b)), are not affected by changes in the ILRT leak testing requirements.

Accident sequences involving containment failure due to severe accident phenomena (group c, above), generally result in large fission product releases from the containment. These are significantly higher than the fission products released due to leakage from isolation system failures. Since these accident sequences are not affected by changes in the leak-tightness requirements of the containment under Appendix J, they can be grouped into a single representative scenario.

Bypass accident sequences in group d (SGTR and Interfacing System LOCAs) are generally not influenced by ILRT testing.

To study the impact of deferring ILRT testing it is only necessary to focus on the accident sequence group for which containment integrity remains intact - group a. With acceptable leakage from containment, some fission products are released to the environment during a core damage accident. Failure to maintain a leak-tight seal could lead to greater fission product leakage from containment. In the Callaway IPE, best estimate containment leakage was assumed based on the results of previous ILRTs. Variation of containment leakage was not expressly considered. In the simplified event tree presented here, failures to fully seal associated with LLRT and ILRT are considered.

Simplified Event Tree

Figure 1 presents the simplified CET used to study the impact of deferring the ILRT for Callaway. Below is a listing of the top events considered in the simplified event tree.

Core Damage Sequence - CD

This top event represents the Callaway Core Damage Sequences from the Level I study. Each core damage sequence from the Callaway Plant Response Tree includes containment response characteristics. Because there are over one thousand Plant Response Tree Sequences, they were binned into 81 Plant Damage States to simplify the interface between the Level I and Level II studies. In the Callaway IPE, failures contained within the cutsets associated with the Plant Damage States were used when assessing dependent failures and branching in the Callaway CET.

Containment Not Bypassed

This top event questions whether containment is bypassed, thus permitting a direct release to the environment. Containment bypass was expressly considered in the Callaway IPE. The bypass sequences include interfacing systems LOCA and steam generator tube rupture (SGTR) sequences which are not isolated. (These sequences are modeled in the Callaway IPE CET top event PSFAIL and are made up of CET end points 47 through 51.)

No Containment Failure from Phenomenon

This top event questions whether containment fails due to severe accident phenomena. The Callaway IPE CET considered a variety of phenomenological containment failure modes associated with severe accidents. Deferring the ILRT will have no impact on these severe accident phenomena. Therefore, all Callaway IPE CET sequences that failed containment as a result of severe accident phenomena (CET and points 2 through 46) can be grouped together.

Containment is Isolated

This top event considers whether the automatic containment isolation systems function after a severe accident. The status of containment isolation was considered in the Callaway IPE plant damage state development. As stated earlier, all Level I core damage sequences were placed in 81 Plant Damage States. These bins were then input into the Containment Event Tree .

After elimination of bypass sequences, the PDS grouping logic diagram top event, containment isolation status, segregates the Level I sequences into sequence groups based on the status of containment isolation at the time of core damage. The loss of isolation branch also includes sequences with containment failure prior to core damage. With the containment not isolated, early and relatively large releases of radionuclides from the plant are possible. If the containment is not isolated the most important additional system consideration from the standpoint of the radionuclide source term is whether the containment sprays function. Consequently, for sequences which are not isolated, this is the only other grouping parameter which is considered. Hence all sequences with containment isolation failure are grouped into two PDSs; isolation failure with spray (PDS #1) or isolation failure without spray (PDS #2).

Failure of containment isolation was not questioned in the Plant Response Trees in order to make the trees tractable in size. Failure of containment isolation at Callaway is generally independent of the other systems modeled in the PRA, including AC and DC power. The probability and contributors to failure of containment isolation were considered separately from the Plant Response Trees. The isolation function failure cutsets were developed from a separate fault tree ensuring that all existing dependencies were modeled consistent with the plant response tree

models. In order to find the probability of PDS #1 and 2 while ensuring that any system dependencies were accurately assessed, the <u>total</u> core damage equation was merged (``ANDed'') with the fault tree for failure of containment isolation. Then, the Boolean equation for containment sprays was merged (``ANDed'') with the previous equation to get the split between PDS bins #1 and #2. The ``ANDed'' equation with the total core damage equation resulted in all of the cutsets being below the 1.0E-10 truncation level due to the relative independence of the containment isolation system. Therefore, all of the sequence frequency is placed in the isolated category and was not considered in the Callaway CET.

In Reference 1, twenty PWR plant IPE submittals were reviewed. While ten submittals reported containment isolation failures as unique accident sequences or plant damage states, the others (like Callaway) did not report isolation failures as unique plant damage states. This was due to the fact that containment isolation failure was negligible. From this review, we can conclude that Callaway's isolation system design is among the most robust.

<u>No LLRT Isolation Failures</u>

While the Callaway IPE considered containment isolation in the PDS development, it did not specifically address whether valves subject to LLRT failed to be leak-tight. This top event considers those random isolation failures during a severe accident where tested containment penetrations fail to fully seal. It should be noted that these failures are not dependent upon core damage sequence and therefore can be treated independently. Since the LLRT schedule remains the same, this top event is not impacted by delaying the ILRT.

<u>No ILRT Isolation Failures</u>

This top event models those failures to seal that are normally detectable by the ILRT. Like LLRT, this top event was not considered in the Callaway IPE and is independent of the core damage sequences sorted in the Callaway CET.

Actual plant experience has shown that almost all ILRT asfound leakage is attributable to leakage routinely identified and corrected through the LLRT program. Since the LLRT schedule remains the same, the simplified model assumes that delaying the ILRT impacts this top event.

<u>Consequence Level</u>

This top event permits the size of containment leakage to vary as a result of failures to seal that would be detectable by the ILRT. In the Callaway IPE, leakage equivalent to 0.04 wt%/day was modeled in the MAAP code to develop the intact containment source term. This value represented the actual Callaway

containment leakage measured during an ILRT performed in 1987. It is considered a best estimate leakage value. In the IPE, 100% of those sequences that resulted in core damage and did not fail containment had this leakage rate. Additional leakage will now be considered for the purposes of studying the deferral of ILRT testing as follows:

Leakage Rate	Amount (wt%/day)	IPE Percentage	Revised Percentage
Best Estimate	0.04	100%	0%
Low	0.2	0%	80%
High	0.4	0%	20%

The revised percentages are based on engineering judgement and are consistent with other leakage assumptions made in Reference 1.

Event Tree Sequences

The simplified CET has seven sequences. However, not all these sequences are impacted by deferring the ILRT. Sequences 6 and 7 are associated with containment failures due to severe accident phenomena and bypass, respectively. These conditions result in large releases and are not impacted by ILRT deferral. Because we are not deferring LLRT, sequence 5 which is associated with LLRT failures to seal remains unchanged. This failure was assumed to be negligible in the original IPE. This assumption will continue to remain valid because ILRT tests to date often identify LLRT-found leakage and include LLRT leakage in their total calculated leakage. By definition, containment leakage not ILRT detectable, sequence 2, is unchanged by the ILRT deferral. This value was assumed to be negligible in the Callaway IPE and again this assumption remains valid. Therefore, only sequence 1 (best estimate containment leakage), sequence 3 (ILRT detectable failure-high consequence), and sequence 4 (ILRT detectable failure - low consequence) are impacted by ILRT deferral.

Results

Table 1 shows the impact of ILRT deferral using the simplified event tree. In the original Callaway IPE, the value for containment isolation was 1.0 making any release from containment due to best estimate leakage. The ILRT-deferred case assumes the top event, containment is isolated, equals 0.0 and uses the previously described percentages for the consequence level. As shown, the overall impact of this redistribution is limited to leakage only sequences. No large source term releases are impacted. To determine the overall risk of this relocation, one has to evaluate the source term associated with the new release paths.

Source Term Evaluation

To evaluate the potential source term increase due to delaying the ILRT, the Callaway IPE Level 2 analyses were used.

Radionuclide Release Characterization

The end points of the CETs (Figure 2) represent the outcomes of possible in-containment accident progression sequences. These endpoints represent complete severe accident sequences from initiating event to release of radionuclides to the environment. The Level I system information was passed through to the containment evaluation in discrete Plant Damage States. An environmental source term could have been associated with each of these containment sequences. However, because of the large number of CET sequences and because of similarities in the sequence characteristics, it was not necessary to develop a source term estimate for each containment sequence. Accident sequences with similar characteristics were therefore grouped into release categories (also called source term categories) to reflect similar release characteristics.

Source Term Category Grouping Parameters

The approach to the definition of source term categories consists of construction of a logic diagram with the grouping criteria defined below, as headings. The end points on the logic diagram represent unique source term categories with their individual characteristics defined by the pathway through the logic diagram.

The first step in the source term assessment effort was to identify the sequence characteristics which are most important to definition of the source term. These characteristics were identifiable from the PDS characteristics and from the CET sequence characteristics since one of the primary objectives in the PDS grouping and CET evaluation was to define those events and conditions most important to source term assessment. This selected set of sequence characteristics important to source term assessment was used as the grouping criteria to define the release categories and the associated source term magnitude, composition, and timing.

The accident sequence characteristics selected for use in the definition of the Callaway source term release categories are:

- Time of release
- Mode of containment failure
- Sprays available and effective in fission product mitigation
- Release reduction factors effective for fission product mitigation

These characteristics were used for the event headings and branch attributes for the source term category (STC) grouping logic

diagram. The STC logic diagram was constructed in approximate order from the earliest, most severe releases (STCs 1, 3, and 5) to the latest, least severe releases (STCs 20 through 24), to no release (STC 25). The general logic used in defining the different release categories is discussed below.

Time of Release (RELTIME)

This release category attribute is considered important because it affects the time available for fission product release mitigation by natural removal processes and spray washout. It also impacts the effectiveness of accident management measures as well as public emergency response measures such as sheltering or evacuation.

The times selected as significant are early and late. Containment failures were treated in the CETs for the relevant sequences. Early containment failure was defined to be at or near the time of reactor vessel failure. In general, early failures would occur during the first day. SGTR sequences, although they are long-term core melt and radionuclide release sequences, were also placed in the early category since the largest portion of the release occurs at or near the time of core melt. It should be recognized, however, that due to the long nature of these sequences, a considerable time exists in which to warn the population and provide for shelter or evacuation. The early category also includes failures such as ISLs, early hydrogen burns, direct containment heating (DCH) events, catastrophic failures, and the vessel thrust failures.

Late containment failures occur at least 24 or more hours after vessel failure, and include late hydrogen burns, long-term overpressure, and basemat melt-through.

The possibility of no containment failure exists and was assigned its own unique source term category.

The releases from containment were dominated by late releases. Early releases only accounted for about 2% while no release represented almost half of the accident sequences.

Mode of Containment Failure (FAILMODE)

This attribute is important because it governs the rate at which fission products are released to the atmosphere. It also affects the magnitude of the release by governing the time available for effective fission product attenuation inside containment.

The attributes considered significant are catastrophic, rupture (early and late), SGTR, ISL, leak (early and late), and basemat penetration. These were evaluated using the branch attributes for the CET headings "Mode of Early Containment Failure" and "Mode of Late Containment Failure" so the definitions were those employed in the CET.

The results are summarized as follows:

FAIL MODE	Frequency (/yr)	Percent of CDF
ISL	1.731E-07	0.33%
Catastrophic	6.974E-09	0.01%
Rupture (early)	3.542E-09	< 0.01%
SGTR	8.559E-07	1.63%
Leak (early)	8.610E-08	0.16%
Rupture (late)	2.849E-08	0.05%
Leak (late)	2.499E-05	47.52%
Basemat	2.778E-06	5.28%
No Release	2.367E-05	45.01%

As shown above, late leak releases were the dominant mode of failure, with basemat about 5 percent and SGTR less than 2 percent.

Spray Available and Effective in Fission Product Mitigation (SPRAYS)

This attribute is considered because it can extend the length of time before late containment failure and it impacts the fission product washout in the containment. The longer the fission products are isolated in the containment atmosphere before release, the more natural mitigation processes (fallout, deposition, etc.) have time to be effective. This attribute also may affect the energy level (temperature) of the release.

The basis for selecting failure or success is the availability of the spray function.

Release Reduction Factors Effective for Fission Product Mitigation

This event indicates several conditions depending upon the containment failure type. For containment failure releases, this attribute is considered to determine whether the release is into and through the auxiliary building, such that the fission products are reduced before exiting to the environment. For SGTR containment bypass sequences, this attribute reflects whether the release is through the MSIVs and the condenser, through a stuck open steam generator safety valve, or whether the release is through a cycling steam generator PORV. For the ISL containment bypass sequences, this attribute reflects whether the interfacing

systems LOCA release area is on the lower level of the auxiliary building, and therefore has a larger reduction in the source term, or is on an upper area with a more direct release path to the environment.

Since most of the releases were either effective or no release, the majority of releases were mitigated to some degree by the release location or resulted in no release to the environment. Note, however, that most of the ISL and SGTR sequences were generally not significantly mitigated.

Source Term Category Results

The results of the source term binning and of the Level II analysis are presented by frequency ranking in Table 2. Figure 3 shows a breakdown of the time and type of containment failure for all sequences. As can be seen, the containment failure modes are dominated by late overpressure failures (predominantly leaks) and no containment failure. Containment bypass sequences (ISL and SGTR) represent less than 2% of the Plant Damage State frequency. Early containment failures, excluding containment bypass scenarios, represent only 0.2% of the total Plant Damage State frequency.

Association of Source Term Categories with Releases

The definition of the source term magnitude, composition, and timing for each release category can be estimated using the following methods:

1. Deterministic analysis of representative sequences from each release category with an accident progression source term assessment code such as MAAP, or

2. By reference to past analysis results such as NUREG-1150, IDCOR, past PRAs, etc.

For the Callaway IPE, method 1 was used. MAAP calculations were performed to assess the source terms directly for 21 of the 25 STCs. For the other four STCs, the release fractions were then characterized by similarity to one of the calculated source terms.

In MAAP, once fission products leave the core in-vessel or core debris ex-vessel, the chemical state is "frozen" and defined by the twelve species listed in Table 3. The chemical state is important in determining the transition between vapor and aerosol forms which affects the deposition and retention of fission products.

Each fission product specie can exist, in MAAP, in up to four states in each region of the containment and of the primary system. These states are "vapor", "aerosol", "deposited", and "contained in the core or in corium". These states, and the

species of Table 3 were used here to characterize the calculated source term characteristics.

Deterministic code calculations using MAAP were used to evaluate source term magnitudes for a wide range of sequences and their variations. Table 4 lists the release fractions of the MAAP "species" for each calculated STC and also identifies the origin of the STC release fractions characterized by similarity. Table 4 also identifies the specific MAAP run used.

As shown in Table 4, four STCs were characterized by similarity with other STCs. These were STCs 3, 4, 10, and 24. STCs 3 and 4 represent the catastrophic containment failure mode which is similar to the early containment rupture failure mode. Since the catastrophic failure mode is highly theoretical, specific MAAP scenarios were not developed. STC 10 represents a SGTR sequence with failure of the MSIVs to close. The releases from containment are expected to be similar to the SGTR case with a stuck open PORV (STC 9). However, the release is through the condenser in this case which will result in a significant reduction in the release fractions due to deposition in the condenser. Conservative decontamination factors (DFs) of 10 for iodines and 100 for the other species (DF of 1 for noble gases) were assumed for this release path over STC 9. STC 24 represents basemat melt-through. This is an extremely long-term scenario and results in a release to the ground below the containment rather than directly to the atmosphere where it can be more easily transported. Based on engineering judgment, a source term for the late containment leakage through the auxiliary building (STC 21) was felt to be a conservative estimate for the source term to be applied in this case. In addition to these four STCs, two other STCs (11 and 25) were defined in a slightly different manner than the rest. STC 11 represents a SGTR without a stuck open PORV and with the MSIVs closed. In this case the PORV will continue to cycle open and closed until the time of vessel breach. Following vessel breach, the PORV will close, effectively isolating the ruptured steam generator. The majority of the release will occur between the onset of core melt and vessel breach. This STC was conservatively characterized by using the release fractions calculated by MAAP at the time of vessel breach.

STC 25 represents no containment failure. This represents 45% of the accident sequences. Although the containment does not fail, minor releases occur due to the allowable Technical Specification limits on leakage. The leakage area modeled in the MAAP code was based upon a 1987 report by Bechtel on the Callaway Containment. This Report concluded that the Callaway containment has a leakage rate of approximately 0.04 wt%/day. The release fractions used were as calculated by MAAP for a typical core melt scenario which did not lead to containment failure. This is conservative since a significant portion of the frequency attributable to this STC represents scenarios in which the core melt is actually arrested in-vessel rather than proceeding to core melt and vessel breach.

Source Terms for Simplified Event Tree

To further quantify the results for the evaluation of ILRT deferral, source terms were generated for the simplified event tree sequences shown in Table 1. The source term categories for the simplified event tree are based upon the Callaway IPE Level 2 analyses. In particular, STC 25 and the approach used to generate it was used in generating simplified event tree source terms. Table 5 provides the release fraction results for the simplified event tree sequences.

Sequences 2 and 5 were determined to be negligably impacted by the ILRT deferral (Table 1) and are therefore not included in Table 5. Sequence 1 release fractions are based upon STC 25 from the Callaway IPE and used a 0.04 wt%/day containment leak rate. This sequence is presented for information only since Table 1 shows that this sequence has a probability of 0.0 for this evaluation.

Source terms for sequences 3 and 4 are new and were generated by MAAP. Sequence 3 represents the high consequence outcome (0.4 wt%/day) for the ILRT detectable failure path and constitutes 9% of the containment failure frequency. Sequence 4 represents the low consequence outcome (0.2 wt%/day) for the ILRT detectable failure path and constitutes 36% of the containment failure frequency. Table 5 presents the MAAP generated release fractions for these sequences.

Sequences 6 and 7 are associated with containment failures due to severe accident phenomena and bypass, respectively. Sequence 6 constitutes 53% of the containment failure frequency and sequence 7 constitutes 2% of the containment failure frequency. Table 5 presents typical release fractions for these sequences obtained from the Callaway IPE.

Conclusion

Table 5 shows a comparison of the release fractions for non-negligible sequences. The worst case release fraction (sequence 6) is 2 to 5 orders of magnitude greater than those of sequences 3 or 4. The largest contributor to containment failure (sequence 6, 53%) has a source term 2 to 5 orders of magnitude greater than the high consequence path for ILRT detectable failures. The release fractions for sequences 1, 3, and 4 are not significantly different. Thus, it has been shown that there is negligible impact on the overall Callaway source term due to ILRT deferral.

The above findings are similiar to the findings of NUREG-1493 for large PWRs. NUREG-1493 concluded there was negligible impact on offsite doses due to increasing the ILRT frequency, given the insensitivity of risk to containment leak rate. Therefore, we conclude that the ILRT deferral will result in no increased risk to the general public.

References

1. EPRI TR-104285, Risk Impact Assessment of Revised Containment Leak Rate Testing Intervals, August 1994.

2. NUREG-1493, Performanced Based Containment Leak Test Program, March 31, 1994 (Draft Revision 2).

Appendix III

Table 1

IMPACT OF ILRT DEFERRAL ON CONTAINMENT RELEASES

Simplified ET Sequence	IPE Probability of Release	ILRT Deferral Probability of Release
1	2.37 E-5 (45%)	0
2	Neg	Neg
3	0.0	4.74E-6 (9%)
4	0.0	1.90E-5 (36%)
5	Neg	Neg
6	2.79E-5 (53%)	2.79E-5 (53%)
7	1.03E-6 (2%)	1.03E-6 (2%)

APPENDIX III

TABLE 2

PERCENTAGE RANKING OF SOURCE TERM CATEGORIES

RANK	SOURCE TERM CATEGORY	FREQUENCY	PERCENT (of 5.259E-05)
1	STC 25	2.367E-05	45.0
2	STC 21	1.585E-05	30.1
3	STC 20	7.914E-06	15.0
4	STC 24	2.778E-06	5.3
5	STC 23	8.155E-07	1.6
6	STC 9	4.382E-07	0.8
7	STC 22	4.071E-07	0.8
8	STC 10	3.749E-07	0.7
9	STC 1	1.645E-07	0.3
10	STC 11	4.280E-08	0.1
11	STC 13	3.416E-08	0.1
12	STC 15	2.327E-08	0.0
13	STC 16	1.762E-08	0.0
14	STC 12	1.705E-08	0.0
15	STC 14	1.162E-08	0.0
16	STC 17	8.816E-09	0.0
17	STC 2	8.656E-09	0.0
18	STC 3	5.315E-09	0.0
19	STC 4	1.659E-09	0.0
20	STC 5	1.367E-09	0.0
21	STC 18	1.367E-09	0.0
22	STC 7	9.935E-10	0.0
23	STC 6	6.842E-10	0.0
24	STC 19	6.840E-10	0.0
25	STC 8	4.971E-10	0.0

APPENDIX III
TABLE 3

MAAP FISSION PRODUCT SPECIES

Specie Number	Specie I.D.	Composition
1	NOBLES	Noble Gases and Radioactively Inert Aerosols
2	CSI	CsI + RbI
3	TEO2	TeO_2
4	SRO	SrO
5	MOO2	MoO_2
6	CSOH	CsOH + RbOH
7	BAO	BaO
8	LA2O3	$La_2O_3 + Pr_2O_3 + Nd_2O_3 + Sm_2O_3 + Y_2O_3$
9	CEO2	CeO_2
10	SB	Sb
11	TE2	Te_2
12	UO2	$UO_2 + NpO_2 + PuO_2$

TABLE 4

COMPOSITE SOURCE TERM CATEGORY RELEASE FRACTIONS

STC	Basis(1)	MAAP Run(2)	Nobles	CSI	SRO	MOO2	CSOH	BAO	LA203	CE02	SB	TE2
1	M	ISL2	0.996	0.406	2.50E-03	0.017	0.414	6.00E-03	2.80E-04	3.00E-03	0.135	0.069
2	M	ISL1	0.996	0.125	1.70E-03	0.014	0.133	3.60E-03	1.90E-04	2.10E-03	0.095	0.053
3	R	(See STC 5)										
4	R	(See STC 7)										
5	M	10	0.996	0.118	7.10E-03	7.00E-03	0.114	4.60E-03	8.70E-04	0.011	0.126	0.106
6	M	10(3)	0.996	0.086	1.50E-03	6.00E-03	0.084	1.90E-03	1.90E-04	2.40E-03	0.059	0.036
7	M	11	0.995	0.111	5.90E-03	5.80E-04	0.113	2.70E-03	5.30E-04	7.90E-03	0.099	0.106
8	M	11(3)	0.994	0.027	1.30E-03	5.30E-04	0.027	6.90E-04	1.20E-04	1.80E-03	0.031	0.031
9	M	SGTR 1B	0.901	0.014	2.20E-03	6.40E-03	0.015	5.60E-03	1.20E-03	4.40E-03	0.087	1.10E-03
10	R	(See STC 9) (4)										
11	M	SGTR 1B(5)	0.865	0.014	2.10E-03	6.40E-03	0.014	5.50E-03	1.20E-03	4.40E-03	9.20E-03	0.000
12	M	10A	0.980	0.064	2.50E-03	1.60E-03	0.067	1.40E-03	3.70E-04	4.40E-03	0.130	0.135
13	M	10A(3)	0.979	0.034	7.10E-04	7.10E-04	0.036	4.50E-04	1.00E-04	1.30E-03	0.063	0.055
14	M	11A	0.913	0.044	1.40E-06	6.70E-05	0.041	1.40E-05	4.40E-08	4.60E-08	3.10E-04	4.40E-06
15	M	11A(3)	0.898	2.50E-03	8.00E-07	3.90E-05	2.30E-03	8.10E-06	2.60E-08	2.60E-08	1.40E-04	5.20E-07
16	M	AD5WDC02	0.954	5.70E-03	2.30E-05	8.90E-07	6.90E-03	1.00E-05	4.90E-06	5.40E-05	0.020	9.00E-03
17	M	AD5WDC02(3)	0.950	3.40E-03	1.60E-05	4.30E-07	4.00E-03	7.40E-05	3.50E-06	3.90E-05	0.014	6.20E-03
18	M	AD5WDC03	1.000	0.012	9.10E-03	9.90E-08	0.012	4.10E-03	7.10E-04	0.016	0.090	0.093
19	M	AD5WDC03(3)	1.000	5.10E-03	3.00E-03	3.90E-08	5.10E-03	1.30E-03	2.40E-04	3.40E-03	0.027	0.032
20	M	AD5WDC04	0.839	3.00E-03	1.90E-05	7.80E-07	3.30E-03	8.60E-06	4.00E-06	4.40E-05	0.017	7.30E-03
21	M	AD5WDC04(3)	0.800	8.10E-04	5.70E-06	1.50E-07	8.50E-04	2.60E-06	1.20E-06	1.40E-05	4.60E-03	2.10E-03
22	M	AD5WDC05	0.958	8.10E-06	3.60E-11	1.50E-10	2.80E-05	3.30E-10	1.20E-12	1.40E-12	9.80E-05	3.30E-04
23	M	AD5WDC05(3)	0.950	5.70E-07	0.000	0.000	1.90E-06	0.000	0.000	0.000	6.90E-06	2.40E-05
24	R	(See STC 21)										
25	M	Case 1(3)	6.40E-04	9.60E-07	4.50E-08	3.00E-08	1.00E-06	3.30E-08	1.60E-09	2.50E-08	4.60E-06	1.20E-05

Notes:

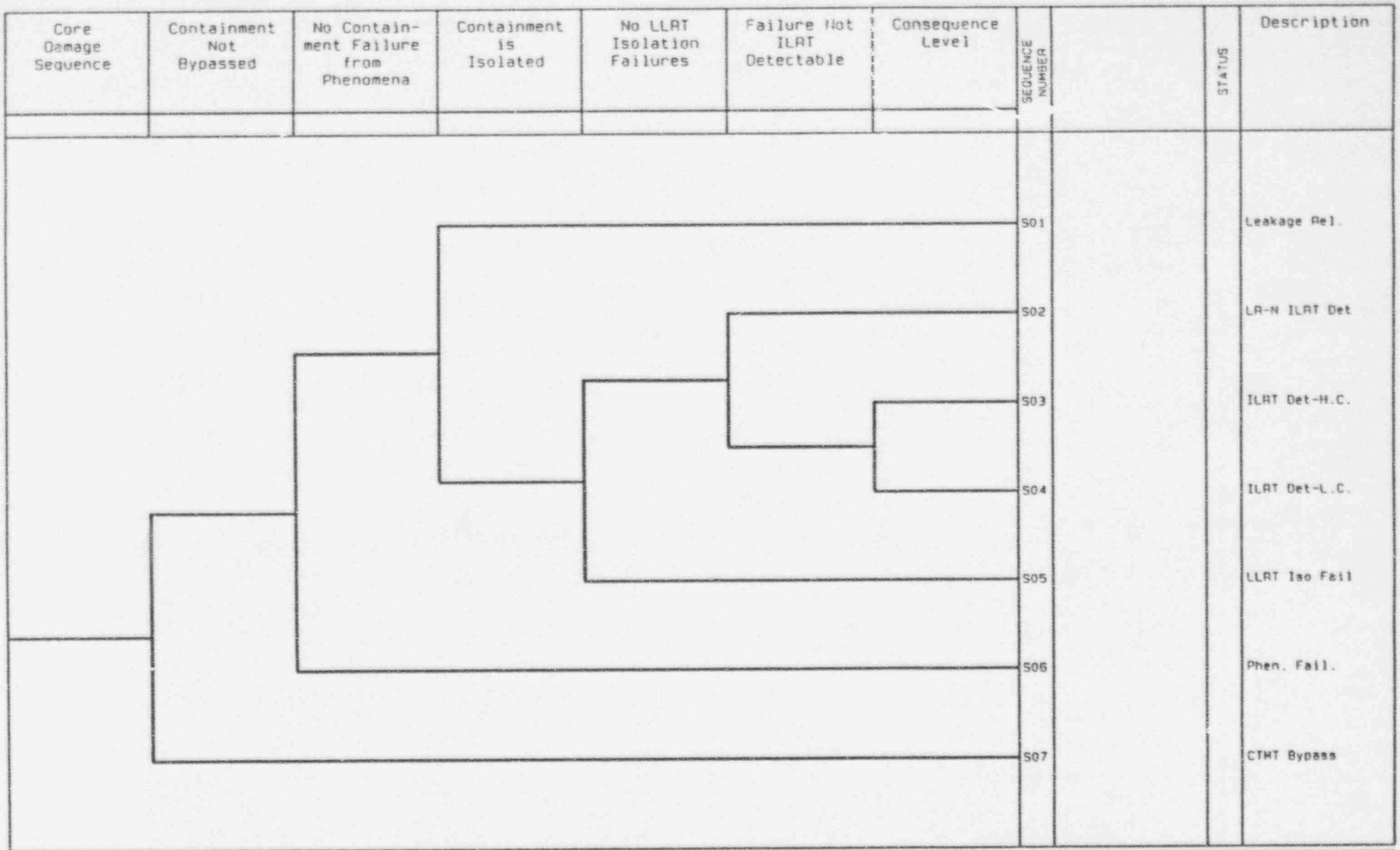
1) Basis: M = MAAP Results; R = Recommended Alternate
2) See Section 4.6 for description of MAAP runs
3) Auxiliary Building Credited
4) Use STC 9 Noble Releases; Reduce CsI by DF of 10; Reduce other species by DF of 100 to account for path through condenser
5) Source term values from SGTR 1B at RV failure

Appendix III

Table 5

Representative Release Fractions for Simplified Event Tree Sequences

Sequence	Nobles	CsI	SrO	MoO_2	C_sOH	BAO	La_2O_3	CeO_2	S_b	Te_2
1	6.4E-4	9.6E-7	4.5E-8	3.0E-8	1.0E-6	3.3E-8	1.6E-9	2.5E-8	4.6E-6	1.2E-5
3	5.3E-3	1.0E-5	3.6E-7	1.7E-7	1.0E-5	1.9E-7	1.7E-8	2.6E-7	1.2E-5	3.9E-5
4	2.5E-3	3.1E-6	1.8E-7	1.1E-7	2.8E-6	1.0E-7	9.3E-9	1.2E-7	1.0E-5	2.7E-5
6	0.996	0.118	7.1E-3	7.0E-3	0.114	4.6E-3	8.7E-4	0.011	0.126	0.106
7	0.996	0.406	2.5E-3	0.017	0.414	6.0E-3	2.8E-4	3.0E-3	0.135	0.069

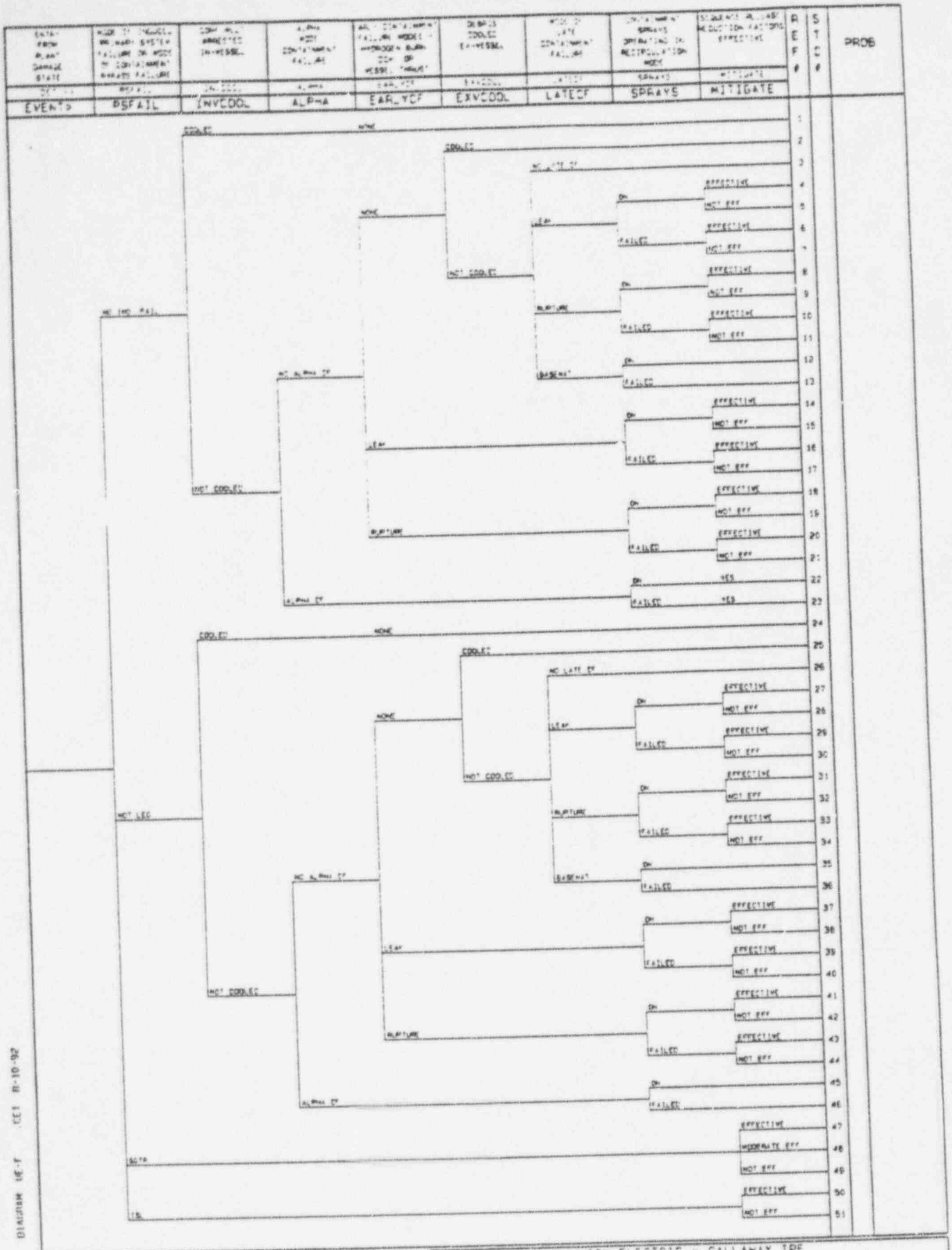


Callaway Plant

ILRT Submittal

Appendix III - Figure 1

Simplified Containment Event Tree



UNION ELECTRIC - CALLAWAY IPE

GENERAL CONTAINMENT EVENT TREE STRUCTURE

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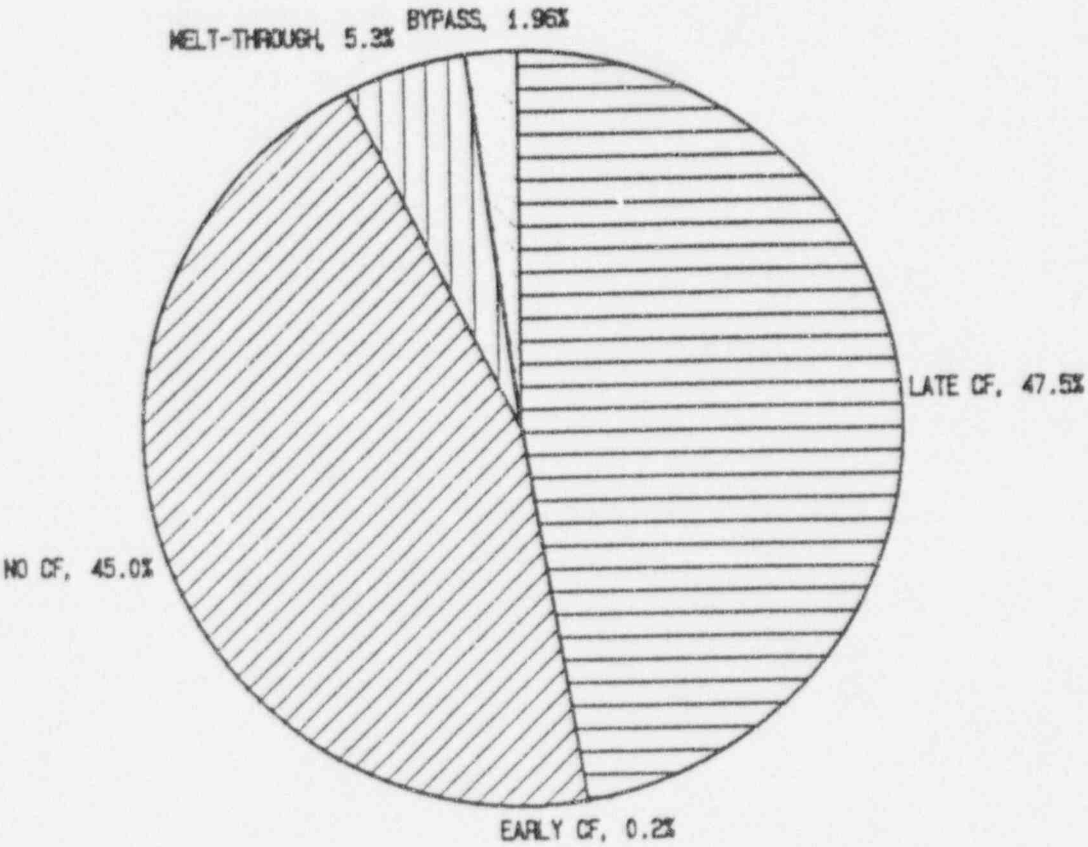
FIGURE 3 CALLAWAY CONTAINMENT FAILURE (CF) MODES



APPENDIX IV - ASSESSMENT OF ILRT BENEFITS AND RISKS

Benefits - LLRT vs. ILRT

The Callaway containment has 81 piping penetrations, 55 electrical penetrations and 10 large special purpose penetrations which include personnel hatches,the fuel transfer tube and spare maintenance penetrations.

Fifteen piping penetrations are associated with water-filled systems designed to support the plant during accident conditions. These include normal charging, high head safety injection, intermediate head safety injection, low head safety injection, essential service water, component cooling water and other safety systems. All of these safeguard systems have internal operating pressures greater than the expected accident pressure within the containment. Therefore, any leakage from these systems would be outward into the containment volume.

Sixteen piping penetrations associated with the secondary side of the Steam Generators are considered part of the containment boundary. Flow through these penetrations, which include the four main feedwater, main steam, blowdown and sample lines, is not interrupted during containment isolation. Piping within these systems located inside containment acts with the containment liner to form the leakage barrier. We consider leakage into these systems from the containment atmosphere to be a low probability event since these systems also operate at higher pressures than the internal pressure expected inside the containment building during an accident.

All remaining piping and electrical penetrations susceptable to leakage are included in the LLRT program. A number of these penetrations contain butterfly valves which provide flow control as well as isolation. Leak tightness is difficult to maintain with butterfly valves and the LLRT program has determined that valves of this type account for most of the as found containment leakage. We utilize LLRTs to trend this leakage and identify the need for valve maintenance or replacement.

For example, ten butterfly valves are used to isolate penetrations of service water piping to and from the containment coolers. During refuel 4 in 1990, LLRTs of these valves revealed significant degradation of valve seals and seating areas. We subsequently replaced these valves with butterfly valves more resistant to such degradation. Thus, a significant improvement in containment boundary integrity was achieved through the LLRT program.

Similarly, all other likely containment leakage contributors are identified through the LLRT program. These penetrations are tested at least once every two years. We rely on the LLRT program to assure containment boundary integrity since leakage through individual penetrations can be accurately identified, measured and corrected. Absent gross degradation of the containment liner and exterior concrete, which has not been observed during visual examinations associated with previous ILRTs, it follows that the LLRT program provides good assurance of compliance with Appendix J requirements.

Aside from containment performance considerations, it should be noted that an ILRT involves expenditures in the range of $2 million and a staff dose commitment of approximately 2 man-rem. In our view, the rigorous and effective LLRT program in place at Callaway diminishes the value of an ILRT and supports our contention that the ILRT scheduled for Refuel 7 can be deferred without negative consequences.

Risks - ILRT Human Error Analysis

In preparation for an ILRT at Callaway, 47 systems are aligned and 50 piping penetrations are vented and drained between containment isolation valves. Vents and drains are left open to ensure that full differential test pressure is experienced across the penetration.

Since 1984, the industry has reported 54 LERs involving ILRT testing. These LERs include the following events, all triggered by human error:

SITUATION	# of LERs
Loss of RHR	2
Inadequate Cold Overpressure Protection	1
Both trains of core spray out of service	1
Contaminated condensate water discharge to the environment	1
Fuel Building vent ESFAS due to actual high activity	1
RPS/ESFAS actuation	4
Valve/Damper out of position	4
Fire watch violations	2
Technical Specification violations	12

The ILRT process is one of the most complex, time consuming and manpower intensive tests that the plant performs. ILRT tests take great planning, attention to detail, training, communication, coordination and supervision to complete the test successfully. Many opportunities are present for human error. During an ILRT, Callaway personnel operate 55 pieces of equipment in 47 systems, perform approximately 20

additional LLRTs and over 172 complex procedure steps. Alignment of containment systems and penetrations requires 1642 valve manipulations. Additional equipment is individually tagged in various operating positions. This includes blocking safety injection and other ECCS signals to preclude inadvertent actuation during the test. All major pumps including Reactor Coolant, Centrifugal Charging, Containment Spray and Safety Injection are tagged out-of-service to eliminate problems during the test. Similarly, major equipment such as Reactor Cavity Cooling Fans, Containment Atmospheric Control Fans, Control Rod Drive Mechanism fans and Containment coolers are also tagged out-of-service.

Minor human error problems have occurred during each of the Callaway ILRTs. For instance, during restoration from the 1990 ILRT, Callaway personnel failed to restore two fuel building radiation monitors. Although these errors were corrected without adverse consequence, the point of this discussion is that exposure to human error increases during complex ILRT operations.

With this in mind, we have attempted to formulate an error probability assessment using the NUREG/CR-1275 THERP prediction estimator. Our analysis indicates that under the best of circumstances, the complications of an ILRT would yield an error-free performance only 70% of the time. Despite the fact that most human errors involved in ILRTs are likely to be inconsequential and easily correctable, there remains a risk of equipment damage, personnel injury and perhaps to nuclear safety which should not be undertaken without achieving commensurate benefits.

We emphasize that our readiness to perform an ILRT incorporates lessons learned from ILRTs at other plants as well as our own experience. We have improved our procedures, enhanced our training, assembled an experienced staff to perform the test, and have improved our equipment performance. We have confidence in our ability to perform an ILRT, but troubling industry events continue to occur because of human performance errors.

Callaway management believes that a key element of our success in nuclear safety, plant reliability and economic performance has been our emphasis on human performance improvements. If we can reduce risk we can enhance our performance. Deferring the ILRT would reduce the risk of human error and allow plant personnel to focus efforts on plant outage restoration and startup -- activities which are key to a safe and successful outage

Conclusion

The ILRT is a complicated, expensive, time consuming test which by its nature can lead to costly human performance errors. The net benefit of an ILRT is confirmation of information already obtained through performance of LLRTs.

Deferral of the ILRT scheduled for Refuel 7 will not involve any unreviewed safety questions or increased risks to the public.