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United States
Nuclear Regulatory Commission

Improved Radiological Consequence Assessment for Dry and Wet Storage of Spent Fuel

Presentation at Water Reactor Safety Meeting

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Consequence Assessment for Dry Storage

Object of the analysis

Provide more realistic quantification, with uncertainty bounds, of offsite doses associated with dry storage cask leakage

Summary of approach

Used RADTRAD code with isotopic inventories for spent fuel after 5 years of decay to calculate individual offsite dose

- Focus of more realistic modeling was aerosol deposition in cask

Conclusion

Modeling aerosol deposition in cask reduces dose by a factor of 400

RADTRAD

Used RADTRAD reactor accident analysis code with isotopic inventories for spent fuel after 5 years of decay to calculate individual offsite dose from dry storage cask leakage

Major RADTRAD models

- **fission product transport and deposition inside a structure**
- **individual offsite dose**

Major RADTRAD inputs

- **fission product release into structure**
- **control volumes and flow rates**
- **leak rate of structure**

RADTRAD

RADTRAD version 3.01 (NUREG/CR-6604, Supp.1) issued June 1999

Approximately 30 industry organizations have requested and received the code

Version 3.02 released April 2000

Code corrections
GUI modifications/improvements

User group

WEBSITE: http://www.nrc.gov/RES/RADTRAD

Dose Modeling for Individual Offsite Dose from Dry Storage Cask Leakage
(From HI-STORM Safety Analysis Report)

Parameter	Value of parameter for...			
	Accident		Off-normal	Normal
Fraction of crud released	1		.15	.15
Fraction of fuel assemblies releasing fission products	1		.1	.01
Fraction of fission product inventory released from each fuel assembly	fission product gas	.3	same	same
	volatile fission products	$2x10^{-4}$	same	same
	actinides, non-volatile fission products (fines)	$3x10^{-5}$	same	same
free volume of the confinement	$6.0x10^{6}$ cm^3		same	same
leak rate of the confinement	$1.3x10^{-5}$ cm^3/sec		$9.5x10^{-6}$ cm^3/sec	$9.5x10^{-6}$ cm^3/sec
dilution factor (i.e., X/Q)	$8.0x10^{-3}$ sec/m^3		$1.6x10^{-4}$ sec/m^3	$1.6x10^{-4}$ sec/m^3
breathing rate	$3.3x10^{-4}$ m^3/sec		same	same
dose conversion factors	Federal Report Guidance Reports 11 and 12 using most conservative clearance class		same	same
accident duration	30 days		1 year	1 year
dose limit	TEDE	5 rem	25 mrem	25 mrem
	thyroid	50 rem	75 mrem	75 mrem
	other critical organ	50 rem	25 mrem	25 mrem
	lense of the eye	15 rem	-	-
	skin dose	50 rem	-	-

Benchmark Calculation

Using parameter values from HI-STORM SAR, ran RADTRAD without modeling deposition to benchmark RADTRAD against current design basis licensing calculation.

Accident Dose Results of Cases 1 and 1nd (mrem)

Organ	RADTRAD Case 1 (with radioactive decay)	RADTRAD Case 1nd (without radioactive decay)	Spreadsheet in HI-STORM Safety Analysis Report
gonads	8.1	8.3	8.3
breast	4.6	4.6	4.6
lungs	120	120	120
red marrow	42	42	42
bone surface	470	470	470
thyroid	4.1	4.1	4.1
remainder	26	26	Not reported
effective (TEDE)	**44**	**44**	**44**
skin	.17	.17	.11

Deposition in Cask

Deposition is modeled in RADTRAD as follows:

$$N = N_0 \cdot e^{-\lambda \cdot t}$$

where N **is the amount of aerosol airborne at time** t **after release,**
N_0 **is the initial amount of aerosol airborne, and**
λ **is the first order rate constant for aerosol deposition**

Approaches to Determine λ

NUREG/CR-6189

- provides rough estimate of λ
- based on dimensions and T-H conditions for a reactor containment

NUREG/CR-6189 adjusted for cask dimensions

- provide insight into effect of dimensions (containment vs. cask)
- not directly applicable, because do not know how much deposition due to each mechanism

Stand-alone calculation of settling using distributions for aerosol density, diameter, and shape factor from reactor accident studies

- ignores additional deposition due to thermophoresis

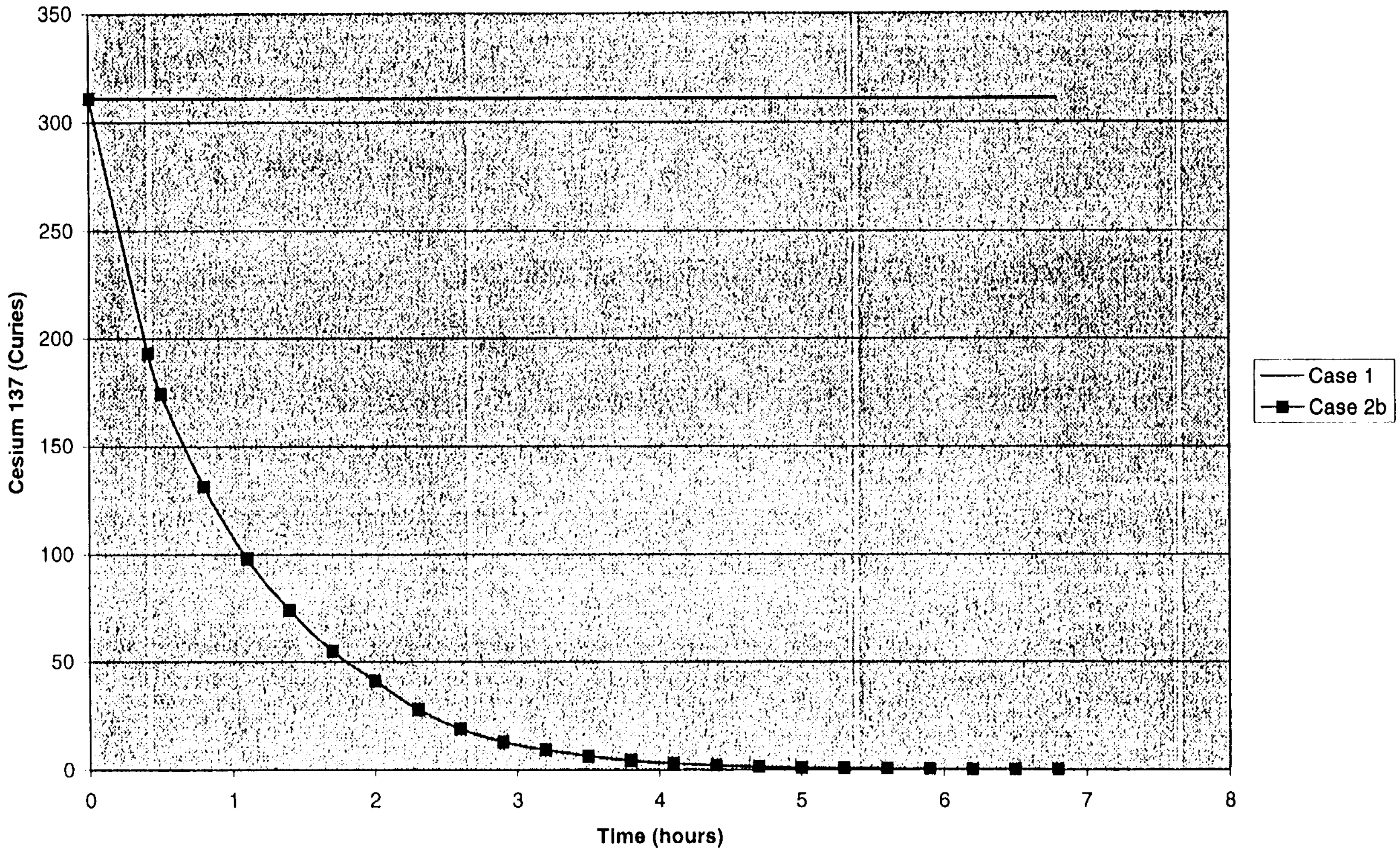
Stand-alone calculation of settling using distributions for aerosol density, diameter, and shape factor for a spent fuel cask

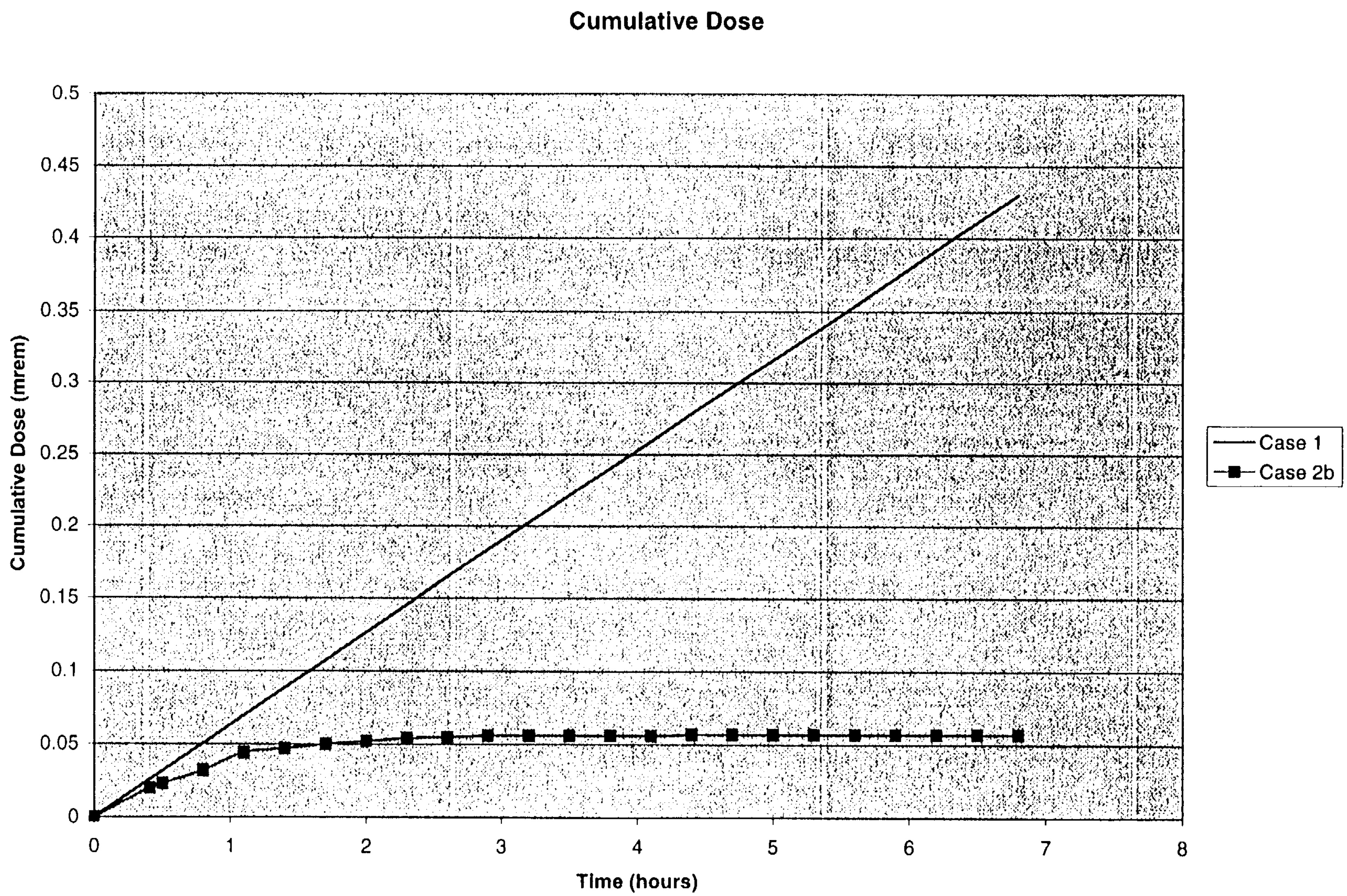
- provides best estimate of λ

Doses for Accident Conditions

Case	Deposition Modeling	TEDE for an Individual at the Site Boundary (mrem)		
		lower bound	best estimate	upper bound
1	none	N/A	N/A	44
2a,2b,2c	NUREG/CR-6189	.037	.059	.097
3a,3b,3c	NUREG/CR-6189 with cask dimensions	.0088	.014	.024
4a,4b,4c	settling only, based on reactor containment conditions	.027	.077	.35
5a,5b,5c	settling only, based on cask conditions	.031	.096	.24

Airborne Particulate in Confinement




Cumulative Dose
Cumulative Dose (mrem)
0.5
0.45
0.4
0.35
0.3
0.25
0.2
0.15
0.1
0.05
0
0
1
2
3
4
5
6
7
8
Time (hours)
Case 1
Case 2b

Rate Constant (hr^{-1}) - Cases 2 and 3

Correlations for λ for reactor containment developed based on phenomenological models for aerosol deposition mechanisms, including gravitational settling, thermophoresis, diffuseophoresis (NUREG/CR-6189)

Time interval (hr)	NUREG/CR-6189			NUREG/CR-6189 with cask fall height[1]		
	Case 2a (90th percentile)	Case 2b (50th percentile)	Case 2c (10th percentile)	Case 3a (90th percentile)	Case 3b (50th percentile)	Case 3c (10th percentile)
.00 - .50	2.0	1.2	.63	9.3	5.3	2.9
.50 - 2.0	1.4	.95	.54	6.4	4.3	2.5
2.0 - 5.0	1.6	1.3	.91	7.4	5.7	4.1
5.0 - 8.3	1.3	.84	.58	5.8	3.8	2.6
8.3 - 12	1.2	.82	.50	5.6	3.7	2.3
12 - 19	1.2	.80	.46	5.6	3.6	2.1
19 - 24	1.2	.79	.44	5.6	3.6	2.0

[1] Fall height for reactor containment is 21 meters
Fall height for HI-STORM confinement is 4.5 meters

Rate constant (hr^{-1}) - Cases 4 and 5

Rate constant for gravitational settling is given by the following equations:

$$\lambda_s = \frac{u_s \cdot A}{V}$$

$$u_s = \frac{\rho \cdot d_e^2 \cdot g \cdot C_s}{18 \cdot \mu \cdot k}$$

where A = **settling area**
V = **confinement volume**
ρ = **aerosol density**
d_e = **aerosol diameter**
g = **gravitational acceleration**
C_s = **Cunningham slip factor**
μ = **viscosity**
k = **aerosol shape factor**

The uncertain parameters are density, diameter, and shape factor.

Rate constant (hr^{-1}) - Case 4

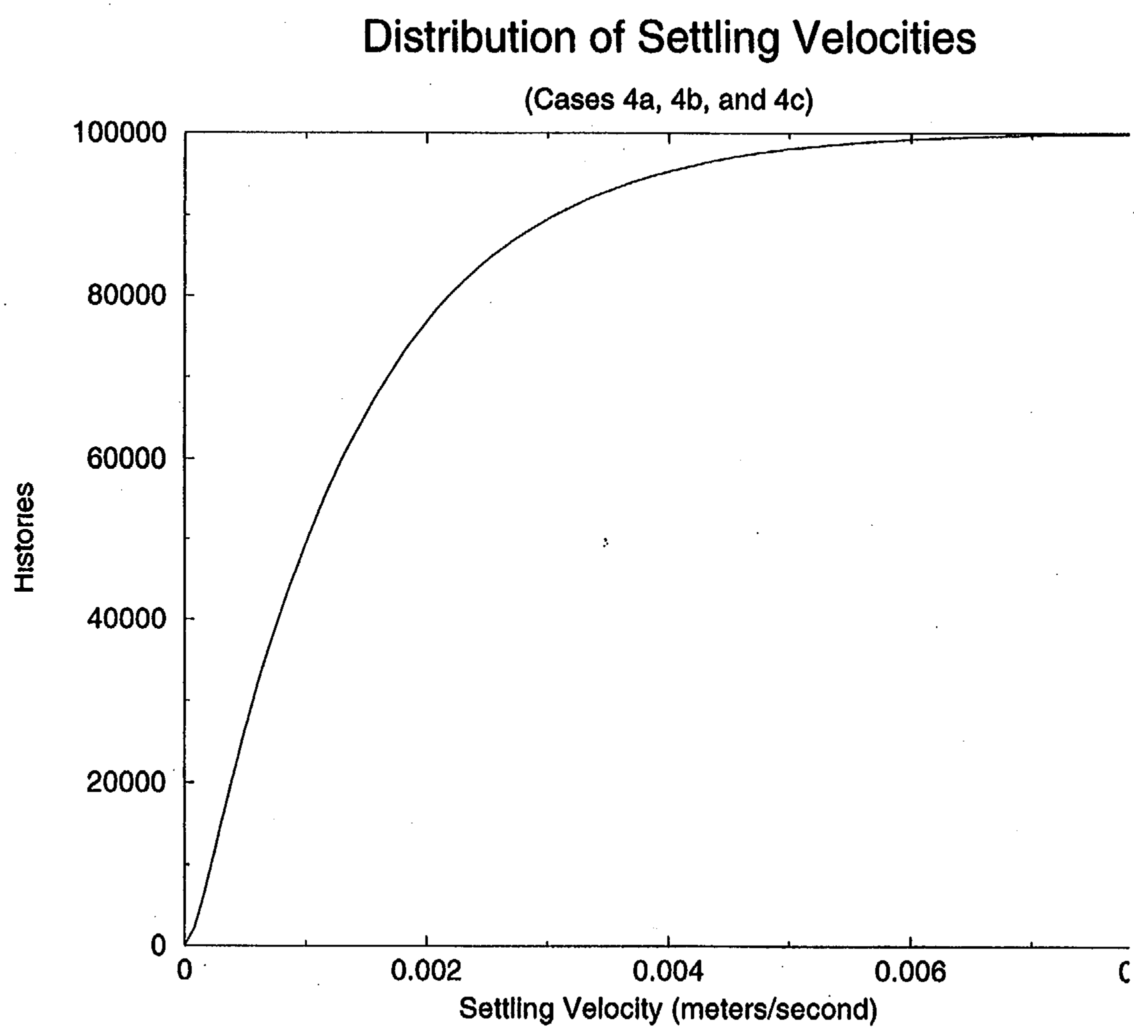
Estimated the rate constant using HI-STORM settling area and confinement volume and distributions for the density, diameter, and shape factor from reactor accident studies (NUREG/CR-6189 and NUREG/CR-5966).

Uncertain Parameter Distributions from Reactor Accident Studies

Parameter	Range of Values	Distribution Type
aerosol density	3.3 to 11 g/cm^3	log-uniform
aerosol diameter	1.5 to 5.5 μm	uniform
aerosol shape factor	1.0 to 4.0	log-normal*

***mean and standard deviation of 1.3 and 3.0, respectively**

Monte Carlo analysis resulted in lower-bound(10th percentile), best-estimate (50th percentile), and upper bound (90th percentile) rate constants of .18, .81, and 2.5 per hour.


Distribution of Settling Velocities
(Cases 4a, 4b, and 4c)
100000
80000
60000
40000
20000
0
Histories
0
0.002
0.004
0.006
Settling Velocity (meters/second)

Rate constant (hr^{-1}) - Case 5

Estimated the rate constant using HI-STORM settling area and confinement volume and distributions for the density, diameter, and shape factor for a spent fuel cask.

Uncertain Parameter Distributions for a Spent Fuel Cask

Parameter	Range of Values	Distribution Type
aerosol density	10 to 11 g/cm^3	uniform
aerosol diameter	1.0 to 4.0 μm	log-normal*
aerosol shape factor	1.0 to 1.3	uniform

***mean and standard deviation of 2.0 and 2.5, respectively**

Monte Carlo analysis resulted in lower-bound(10^{th} percentile), best-estimate (50^{th} percentile), and upper bound (90^{th} percentile) rate constants of .25, .65, and 2.1 per hour.

Doses for Normal Conditions

Case	Deposition Modeling	TEDE for an Individual at the Site Boundary (mrem)		
		lower bound	best estimate	upper bound
1	none	N/A	N/A	.42
2a,2b,2c	NUREG/CR-6189	3.2E-5	5.1E-5	8.2E-5
3a,3b,3c	NUREG/CR-6189 with cask dimensions	9.4E-6	1.4E-5	2.2E-5
4a,4b,4c	settling only, based on reactor containment conditions	2.4E-5	6.6E-5	2.9E-4
5a,5b,5c	settling only, based on cask conditions	2.8E-5	8.1E-5	2.0E-4

Comparison with MELCOR Results

MELCOR accident analyses performed for TN-125 cask with a 4 mm^2 hole (SAND98-1171/7, *Data and Methods for the Assessment of the Risks Associated with Maritime Transport of Radioactive Materials, Results of the SeaRAM Program Studies*, May 1998).

Calculated accident dose using RADTRAD for HI-STORM using MELCOR-predicted deposition rate constants from the TN-125 cask study.

RADTRAD accident doses using MELCOR-predicted deposition rate constants were .070 to .11 mrem.

RADTRAD accident dose using settling rate constant (Case 5b) was .096 mrem.

Excellent agreement because settling was dominant mechanism in MELCOR analyses.

Effect of Aerosol Concentration

Effect was examined in the MELCOR analyses of TN-125 cask

Increasing fuel fines release fraction by a factor of 100 ($1\text{x}10^{-5}$ to $1\text{x}10^{-3}$) increased the deposition rate constant by a factor of 3, due to increased coagulation.

Cases 2a to 4c based on aerosol concentrations in reactor containment

Containment concentration: ~ 7 g/m^3

Dry cask concentration:	accident conditions	~ 70 g/m^3
	normal conditions	~ .7 g/m^3

Accident doses calculated for Cases 2a to 4c would be a little lower if the higher cask aerosol concentration were considered.

Normal doses calculated for Cases 2a to 4c would be a little higher if the lower cask aerosol concentration were considered.

Effect of Volatile Isotopes

ISG-5 includes release fraction of 2×10^{-4} for class of fission products called "volatiles." This class consists of cesium, ruthenium, and strontium.

RADTRAD analysis assumes volatiles are in aerosol form.

Subsequently estimated the amount of volatiles in vapor form under accident conditions and normal conditions.

Based on this estimate concluded:

> **For accident conditions, the fraction estimated to be in vapor form is expected to have a small effect on the accident dose.**
>
> **For normal conditions, the fraction estimated to be in vapor form may result in the normal dose being up to 1.2×10^{-3} mrem, which is a factor-of-15 higher than RADTRAD Case 5b (8.1×10^{-5} mrem).**

Comparison with Direct Shine Dose

Pathway	TEDE (mrem)	
	Accident Conditions (30 days)	Normal Conditions (365 days)
Direct Shine[1]	60	700
Leakage	44	<4.2[2]
Leakage with deposition	.096	<.012[2]

[1] Direct shine dose is for 2x5 array at a distance of 100 meters (Figure 5.1.3 of HI-STORM SAR).

[2] Dose is for 10 casks.

Possible Future Work

Quantify dose reduction from deposition mechanisms other than settling.

Surface	Area (m^2)	Orientation
confinement floor	2.37	upward facing
confinement ceiling	2.37	downward facing
confinement wall	24.7	vertical
basket	189.6	vertical
outside of fuel channel	149.5	vertical
inside of fuel channel	149.5	vertical
fuel cladding	680.0	vertical

Estimate the uncertainty in offsite dose due to variability in the weather.

Conclusions

More realistic leakage doses for one cask under accident and normal conditions are .096 and .0012 mrem, respectively.

Most fission products deposit within cask in a couple of hours.

Neglecting aerosol deposition increases offsite dose by about a factor of 400.

NRC is considering modifying RADTRAD to allow production use for dry storage cask leakage analysis.

Research Perspectives on the Evaluation of Steam Generator Tube Integrity

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Abstract

Industry efforts have been largely successful in managing degradation of steam generator tubes due to wastage, pitting, and denting, but fretting, SCC and intergranular attack have proved more difficult to manage. Although steam generator replacements are proceeding there is substantial industry interest in operating with degraded steam generators, and significant numbers of plants will continue to do so. In most cases degradation of steam generator tubing by stress corrosion cracking is still managed by "plug or repair on detection," because current NDE techniques for characterization of flaws are not accurate enough to permit continued operation. This paper reviews some of the historical background that underlies current steam generator degradation management strategies and outlines some of the additional research that must be done to provide more effective management of degradation in current generators and provide greater assurance of satisfactory performance in replacement steam generators.

1. Introduction

Steam generators have historically been among the most troublesome of the major components in commercial pressurized water reactor (PWR) nuclear power plants around the world. They have been described as the "weak link"[1] in the PWR design, and their premature deterioration has been characterized as "one of the most persistent challenges facing utilities with pressurized water reactors."[2] For the past decade or more, steam generator problems in the United States have ranked only behind refueling outages as the most significant contributor to lost power generation.[3] Beyond the reliability and economic issues facing utilities, steam generator problems raise potentially significant regulatory issues within the U.S. Nuclear Regulatory Commission.

The magnitude of the steam generator tube degradation problem is illustrated by the fact that, to date, various forms of degradation have resulted in the plugging of more than 100,000 tubes worldwide. In 1998, the last year for which complete data are available, 45% of 230 operating PWRs in the EPRI survey of steam generator performance were required to plug tubes. In addition, 150 steam generators in 51 PWRs around the world had been replaced because of severe tube degradation, including 68 steam generators in 22 U.S. plants.[3] Replacements are continuing in both the U.S. and abroad.

The present paper discusses research perspectives related to steam generator tube degradation, with emphasis on problems associated with stress corrosion cracking (SCC). Such SCC can be either axial and circumferential in orientation and can occur at various locations in steam generators, initiating at both the inner and outer surfaces of the tubes. The history of steam generator tube degradation is briefly reviewed, and the evolution of technologies for the nondestructive examination of steam generator tubes is summarized. Also considered are the effects of flaws and flaw morphologies on the structural and leak integrity of steam generator tubing, and the difficulties in detecting, characterizing, and analyzing the structural effects associated with the more complex crack geometries observed in recent years is discussed. The potential for tube failure under severe-accident conditions is also considered.

2. Corrosion Degradation of Steam Generator Tubes

Corrosion problems have afflicted steam generators from the very introduction of the PWR technology. Shippingport, the first commercial PWR operated in the United States, developed leaking cracks in two Type 304 stainless steel (SS) steam generator tubes as early as 1957, after only 150 h of full-power operation.[4-6] The cracks were attributed to stress corrosion cracking (SCC) produced by free caustic in the secondary water and steam blanketing of the tubes at the top inlet portion of the steam generator, leading to concentration of the caustic. The leaking tubes were plugged, and the use of a modified sodium phosphate water chemistry was instituted to control secondary water pH.

Because austenitic SS steam generator tubes were found to be susceptible to SCC from both chlorides and free caustic, the decision was made in the late 1960s to instead use Alloy 600 tubes in the United States and most of Europe and Alloy 800 tubes in Germany.[1] The decision to use Alloy 600 was made on the basis of its known high resistance to chloride attack, based largely on petrochemical plant experience. However, it ignored the work of Coriou et al.[7,8] which showed as long ago as 1959 that this and similar nickel-base alloys were subject to stress corrosion cracking in deionized water at 300-350°C. Alloy 600 steam generator tubes have undergone a series of successive failure modes since that time.

Because Alloy 600 is subject to cracking at high caustic concentrations, early steam generators with Alloy 600 tubes used phosphate additions to the secondary water to provide a buffering capability. This was based on prior experience with fossil-fired boilers. However, rapid caustic cracking was observed in several early steam generators. This problem was successfully controlled by reducing the sodium-to-phosphate molar ratio, but severe tube wastage problems were almost immediately experienced. By the early-to-mid 1970's, wastage was by far the leading cause of tube plugging in the U.S.

In response to the tube wastage problem, most U.S. plants switched to all volatile water treatment (AVT) around 1974. In this approach, ammonia, morpholine or similar additions were added to control pH and hydrazine or similar additions were added for oxygen scavenging. Effective use of AVT water chemistry requires very high purity levels in the feedwater, since no buffering agents are present to prevent excessive acidic or caustic conditions in regions of impurity concentration. It should be noted that once-through steam generators have always used AVT water chemistry to avoid the deposition of chemical solids on tube surfaces in their boil-dry design.

The wide-spread change to AVT water chemistry resulted in a dramatic decrease in tube plugging due to wastage, but this problem was soon replaced by severe tube denting problems in many plants. Tube denting was eventually brought under control by improved controls on feedwater chemistry, improved condenser integrity to eliminate the inleakage of oxygen and other impurities, and, in some cases, the use of condensate polishers or boric acid additions.

Since about 1980, steam generator tube degradation in the U.S, and elsewhere has been dominated by stress corrosion cracking. Unlike wastage and denting, which occur exclusively on the secondary side (outer diameter) of the tubes, stress corrosion cracking can occur on either the primary or secondary side. Primary water stress corrosion cracking (PWSCC) is most likely to occur at regions of high residual stress, as at the tube expansion transition at and immediately above the tubesheet, at U-bends (particularly the small-radius U-bends on the inner-row tubes, and in tube regions deformed by secondary–side denting at the tube support plates.

A number of design changes have been implemented over the years to address the PWSCC issue. These include shot peening or rotopeening of the ID surfaces of the tubes in the roll transition zone to produce compressive residual stresses, improved processes for expanding the tubes into the tubesheet to reduce residual stresses in this region, and in–situ thermal treatment of U-bends on older plants and thermally treated U–bends in newer plants to reduce residual stresses in this region. All of these processes have proven at least somewhat beneficial, but PWSCC continues to be a problem in PWR steam generators.

Outer-diameter stress corrosion cracking (ODSCC) and intergranular attack (IGA) commonly occur in crevices or under corrosion product scales, where conditions are such that incomplete wetting by secondary water occurs, and the consequent alternate wetting and drying result in substantial local buildup of corrosive species. Such locations include the tube support plate crevice, the region near the top of the tube sheet, free span areas under corrosion products or deposits, and regions under sludge build–up. Calculations of local crevice chemistry predict concentration factors approaching 10^8 and crevice pH values ranging from < 2 to > 10 at operating temperatures, depending upon the impurity species in the secondary water. Again, remedial actions have been taken over the years to address this problem. However, ODSCC, like PWSCC, continues to be a leading cause of steam generator tube plugging and repair.

In the 1980s, PWSCC and ODSCC problems were almost entirely confined to low-temperature mill annealed (LTMA) tubing found in Westinghouse steam generators. However, starting about 1990, SCC problems also began to significantly affect the high-temperature mill annealed (HTMA) tubes used in the Combustion Engineering steam generators.[3,9,10] More recently, SCC is increasingly observed in Babcock & Wilcox steam generators, which use tubes that have been stress relieved (SR) tubes after a similar high-temperature 1065-1090°C mill anneal.[3]

Around 1980 for replacement units and in the mid-1980s for the new Model D-5 and Model F steam generators, Westinghouse began using thermally treated (TT) Alloy 600 tubing, which has a microstructure more resistant to SCC, and the oldest of these units (e.g., Surry 1 and 2) have operated for 20 years with virtually no SCC.[3] However, numerous SCC failures have been observed in Alloy 600 TT mechanical plugs, an effect attributed to certain susceptible heats of material,[11–17] and PWSCC of the Alloy 600 TT tubes in the Ulchin 1 and 2 steam generators in Korea has led to extensive tube sleeving, and replacement of these steam generators is now under consideration.[18–20]

Since about 1989, thermally treated Alloy 690 has been the tubing material of choice for replacement steam generators. After up to 11 years of service, no incidents of SCC have been reported for any of these tubes in the field. While laboratory studies have also been unable to produce SCC in Alloy 690 in primary water chemistries, numerous studies have demonstrated the ability to crack this alloy under conditions that approximate the water chemistries and impurities expected in steam generator crevices.[21–24]

3. Steam Generator Integrity

3.1 Nondestructive Evaluation

To be able to ensure the integrity of steam generator tubing, it is important to be able to detect and characterize the degradation. Up to the early 70s the inservice inspection of PWR steam generators was carried using single-frequency eddy current (EC) bobbin coils. This inspection technology was adequate for detection of volumetric degradations but not for cracks. Part of the problem was a low signal to noise ratio for cracks, and in the late 70's, two-frequency EC equipment was introduced to help reduce noise signals from probe wobble and the tube support plate.

By the mid-80s additional modes of degradation such as pitting and intergranular attack (IGA) had to be addressed. Pancake coils were introduced in the 80's to improve detection of IGA in the tube sheet crevice. In addition, three-frequency mixing of bobbin coil signals was introduced to improve flaw detection. Dodd and Deeds[25] showed how to eliminate the tube support plate (TSP) signal by using magnitude and phase in a least-square analysis of data at different frequencies. Steam generator inservice inspection guidelines (ISI) guidelines were introduced by EPRI in the 80's that included qualification requirements for techniques and analysts that focussed on performance with a requirement that the inspector demonstrate an 80% probability of detection (POD) for flaws > 60% throughwall rather than mere adherence to procedures.

By 1990, motorized rotating pancake coils (MRPC) with single or multiple probe heads and isometric displays of the eddy current response were being used to supplement bobbin coil inspections. The 90's saw extensive use of MRPC for better characterization of cracks in U-bends, TSP, and the roll transition zone (RTZ). In addition to the extensive use for supplementary inspections in locations susceptible to cracking, MRPC were used for primary examinations in some cases such as the detection of circumferential cracking. Differential MRPC designs like the +Point probe were introduce to provide improved signal to noise ratios in many cases. Despite improvements in detection capability, sizing, however, is still a problem in many cases.

3.2 Failure Models for Steam Generator Tubes

Extensive work by NRC[26] and industry[27–28] during 70's and 80's has developed and verified models for failures of flawed steam generator tubes under normal operating temperature (300°C) and pressures up to the failure of unflawed tubes (10,000–11,000 psi). Failure of steam generator tubes under such conditions is controlled by the flow stress of the material. A significant body of failure data on flawed steam generator tubes currently exists and has been the basis for the development of various flow stress models.

Most of this work focused on the potential for tube failure during design basis accidents like a main steam line break (MSLB). Risk assessment studies,[29] however, show that a significant portion of the risk due to steam generator tube failures is associated with tube failures due to severe accidents, during which tube temperatures can increase to 650–750°C. Under such conditions the strength of Alloy 600 decreases significantly as shown in Fig. 1, and creep becomes a potential failure mechanism for the tubes and the potential for increased leakage through flaws due to opening of existing throughwall flaws by creep deformation must be considered.

The research program at ANL has developed a data base, correlations and methodologies for predicting the failure of flawed and unflawed steam generator tubes under severe accident conditions.[30] It is well known[31] that high-temperature failure is controlled by thermal creep at low strain rates and by flow stress at high strain rates. In the most structurally challenging severe accidents, the coolant pressure remains high (e.g., close to the safety relief set point) whereas the temperature of the tubes rises at rates varying from 5–10°C/min. Tests have shown that under these conditions, tube failure is best described by a creep model.[30]

Such models can also be extended to consider the potential for the failure of repaired tubes under severe accident conditions. Analyses and tests have been performed to study the behavior of tubes repaired by the Electrosleeve™ process under severe accident conditions.[32]

Stress corrosion cracks on the primary side and due to high caustic concentrations on the secondary are mainly planar in nature. The cracks of primary interest currently, particularly on the secondary side, are segmented and have many ligaments between small segments of the crack. These cracks behave differently structurally from planar cracks. Tubes with these kinds of cracks exhibit higher burst pressures than one would predict using the correlations based on a planar bounding crack.

Mechanistic approaches to prediction of failure require the characterization of flaw geometries. In some cases it is possible to develop empirically based Alternate Repair Criteria (ARC) that do not explicitly consider flaw geometry. For example, in the case of ODSCC in Westinghouse steam generators with drilled-hole tube support plates, Generic Letter 95-05 provides repair criteria in terms of the peak bobbin coil voltage that do not explicitly address crack length or depth. Such empirical approaches presume that the crack geometries in the tubes used to develop the data base for the empirical correlation are representative of those actually encountered in reactor.

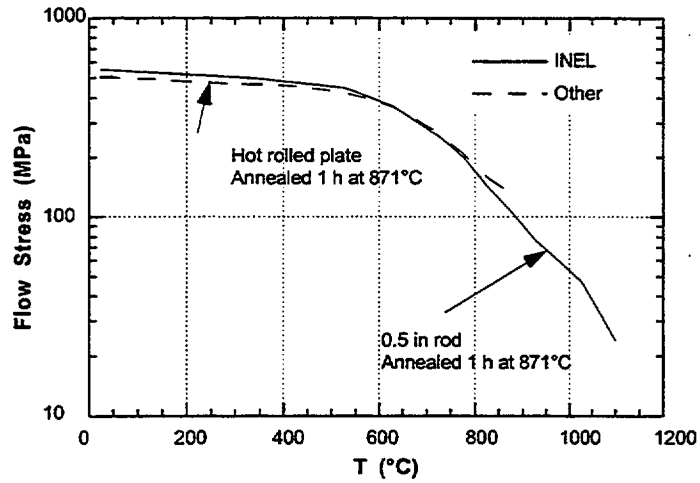


Figure 1.
At high temperatures the flow stress of Alloy 600 decreases markedly and creep effects become significant

4. Regulatory Guidance

Regulatory guidance for steam generator tube plugging and repair was developed in the 1970s (Reg. Guide 1.121, *Bases for Plugging Degraded PWR Steam Generator Tubes*). This guidance was based on deterministic depth based plugging criteria that did, however, attempt to account for degradation growth and NDE uncertainty. The specific estimates for degradation growth and NDE uncertainty in Reg. Guide 1.121 were based on engineering judgment. Because the uncertainties associated with the integrity assessments are strongly dependent on the specific mode of degradation, regulatory guidance has changed with time from deterministic depth-based plugging criteria that apply to all flaw types toward performance-based risk-informed degradation-specific plugging criteria. One example of such guidance is Generic Letter 95–05, which as noted previously, provides plugging and repair criteria for a specific degradation, ODSCC at tube support plates in Westinghouse steam generators with drilled–hole tube support plates. The NRC staff has considered more broadly applicable performance-based risk-informed guidance in DG–1074, *Steam Generator Tube Integrity*, and such an approach is reflected in current industry guidance for tube integrity assessments (NEI 97–06).

One major outcome of regulatory activity over the past 10 years to develop guidance for tube integrity assessments is the development and implementation of two key concepts, condition monitoring and operational assessment. Condition monitoring is an assessment of the current state of the steam generator relative to the performance criteria of structural integrity. An operational assessment is an attempt to assess what will be the state of generator relative to the structural integrity performance criteria at the end of the next inspection cycle. The predictions of the operational assessment from the previous cycle can be compared with the results of the condition monitoring assessment to verify the adequacy of the methods and data used to perform the operational assessment.

The reliability of such assessments and projections depends critically on the reliability of the NDE techniques used to establish the flaw distribution both in terms of detection of flaws and characterization of flaws, the capability to assess the impact of these flaws on the structural integrity of the tubes, and the ability to predict crack initiation, evolution, and growth. In most cases degradation of steam generator tubing by stress corrosion cracking is still managed by "plug or repair on detection," because current NDE techniques for characterization of flaws are not accurate enough to permit continued operation. This is very conservative in many cases, since flaws less than 40% throughwall or even deeper, short flaws have very little impact on tube integrity. On the other hand, current inspection technologies and procedures can miss flaws that will lead to steam generator tube ruptures.

5. Ongoing and Future Research

5.1 NDE Round Robin

An independent assessment of steam generator inspection reliability is being developed through an NDE round–robin on a steam generator mockup at Argonne National Laboratory. The purpose of the round robin is to assess the current state of SG tubing ISI reliability, determine the probability of detection (POD) as function of flaw size or severity and assess flaw sizing capability. Eleven teams have participated in analyzing bobbin coil and rotating coil mock-up data collected by qualified industry personnel. The mockup contains hundreds of cracks and simulations of the artifacts such as corrosion deposits, tube support plates, etc. that make detection and characterization of cracks more difficult in operating steam generators than in most laboratory situations. An expert group from ISI vendors, utilities, EPRI, ANL, and the NRC have reviewed the signals from the laboratory grown cracks used in the mockup to ensure that they provide reasonable simulations of those obtained from real cracks. The number of tubes inspected and number of teams in the round robin are intended to provide better statistical data on the probability of detection (POD) and characterization accuracy than is currently available from industry performance demonstration programs.

5.2 Advanced NDE

Other current research in EC NDE involves the development of advanced modern imaging and analysis algorithms[34]. Codes have been developed that permit more efficient and flexible analyses of rotating coil data. Simplification of interpretation of data is provided though improved enhanced visualization and automated analysis methodologies. It is now possible to produce NDE profiles of large sections of tubing in a fraction of the time that is needed for manual analysis.

Manual analysis of multiple frequency eddy current data is a tedious and challenging process. Signal distortion by interference from internal/external artifacts in the vicinity of flaw further complicates discriminating of flaw signals from noise. In comparison to high-speed bobbin coil inspections, high-resolution multi-coil rotating and array probes generate enormous amounts of data over comparable scan lengths. Rotating probe ISI of SG tubing is thus generally restricted to areas that are historically predisposed to known damage mechanisms and sections of particular interest that are flagged by the initial bobbin coil examinations. More extensive application of such probes for improving NDE reliability rests in part on automating various stages of the data screening process. Computer-aided data analysis is the only viable way to overcome many of the challenges associated with reliable processing of data acquired with high-resolution probes.

In order to characterize flaws in a SG, a variety of characterization methods are being examined. An automated imaging and analysis algorithm for the analysis of RPC data has been developed. The basic elements of the algorithm include automated calibration of the data, filtering and deconvolution to improve the signal to noise ratio, use of a rule-based expert system to classify indications, and the use of multifrequency, multiparameter correlations for flaw size.

The method also provides a graphical display which helps visualize cracking especially in cases like the roll transition where geometry greatly complicates analysis. The results can be presented directly in terms of depth profiles as a percentage of the tube wall thickness. Reconstruction of helically scanned data into C-scan format allows for the observation of sizing results from any azimuth and elevation view angle and for any axial or circumferential cross section of the tube. Typical examples of the graphical display are shown in Figs. 2a and 2b.

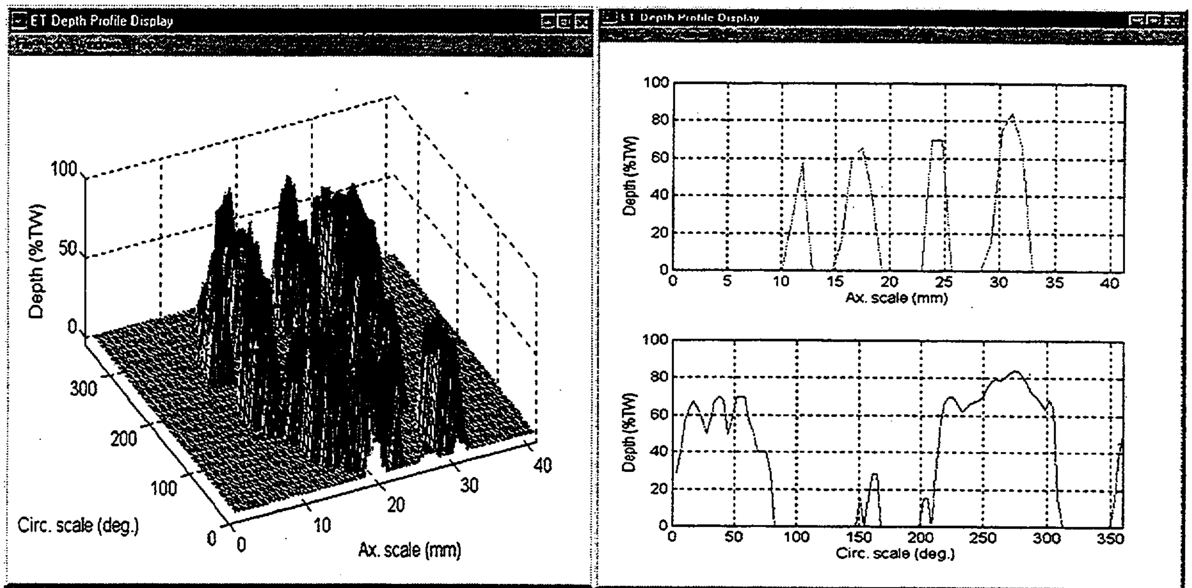
In the development of the multiparameter algorithm the results from the algorithm have been compared to fractographic results on a wide variety of SCC cracks and EDM and laser notches. To provide an objective benchmark, however, additional SCC cracks were produced and used for a blind test of the predictions of the algorithm against fractographic measurements of the crack geometry. Examples of the comparison of the NDE results with fractographic measurements are shown in Fig. 3.

5.3 Structural Integrity

As noted previously, well-established criteria for predicting ligament rupture and unstable burst pressures of tubes with relatively long rectangular flaws exist. Some modifications of these criteria have been made for short and deep flaws based on recent tests at ANL [13]. Although we can currently predict with some confidence failure pressures of tubes with flaws that are rectangular in shape, such a morphology is not characteristic of much of the cracking that is currently being observed in steam generators. Stress corrosion cracks in steam generator tubes are generally non-planar, ligamented, and can have highly complex geometry. Procedures for predicting ligament rupture for such complex cracks, using an "equivalent rectangular crack" approach (Fig. 4), has recently been developed[32,33]. Limited tests at ANL on steam generator tubes with laboratory generated stress corrosion cracks have shown the usefulness of such an approach. The use of the "equivalent rectangular crack" to predict leak rates through laboratory generated stress corrosion cracks has also proven to be promising. Additional leak rate and failure tests on tubes with stress corrosion cracks that are generated in the laboratory as well as on pulled tubes from a retired SG are currently being planned to further validate the approach.

5.4 Materials Degradation

Although the performance of Alloy 690TT material has been excellent to date, further work is needed to characterize its potential for SCC. As stated previously, laboratory studies have identified a variety of environments in which SCC of Alloy 690 TT occurs. It is commonly



(a) (b)

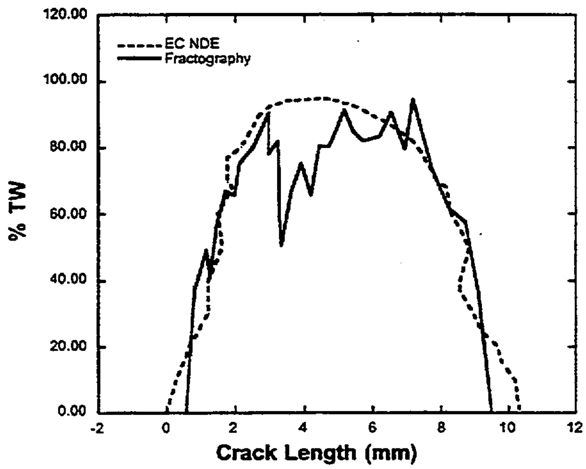
Figure 2. (a) Lab-grown circumferential ODSCC ≈360° staggered cracking with maximum depth > 80% TW. (b) Axial or circumferential cross-sections can be taken to get profiles of the crack

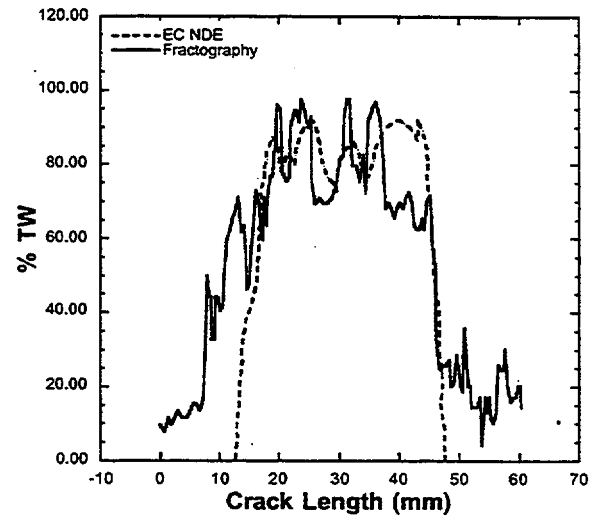
observed in mildly acidic and slightly oxidizing solutions and may be aggravated by the presence of chlorides or sulfates.[35] These mildly acidic conditions are particularly relevant because molar ratio control as well as sea water contamination can produce mildly acidic crevices in steam generators.

It has been observed that concentrations of lead in the ppm range in mildly acidic, neutral, and alkaline environments produce or substantially accelerate the SCC of Alloy 690.[21,36] Despite efforts to reduce lead contamination, it still persists in deposits inside steam generators in ranges substantially greater than the levels required to produce SCC in the laboratory. However, this Pb contamination has not yet produced SCC in the field, and the reasons are unclear.

Sulfur in the form of sulfides is another contaminant that is known to accelerate SCC in Alloy 690 TT, based on experiments in alkaline solutions. Sulfur is sometimes introduced as sulfates as feedwater contamination or in resin fines. These sulfates, in turn, can be reduced to sulfides either by hydrazine or by direct reaction with Alloy 690TT, and this reduction process and its consequences have not been adequately characterized.

Additional research is needed to address these and other issues. The crevice chemistries at various locations in steam generators must be determined, and the chemistry of Pb-containing deposits and their possible relationship to the occurrence of SCC should be evaluated. In addition, the conditions under which lower-valence sulfur compounds can form in crevice environments due to reactions with hydrazine or with Alloy 690 requires further





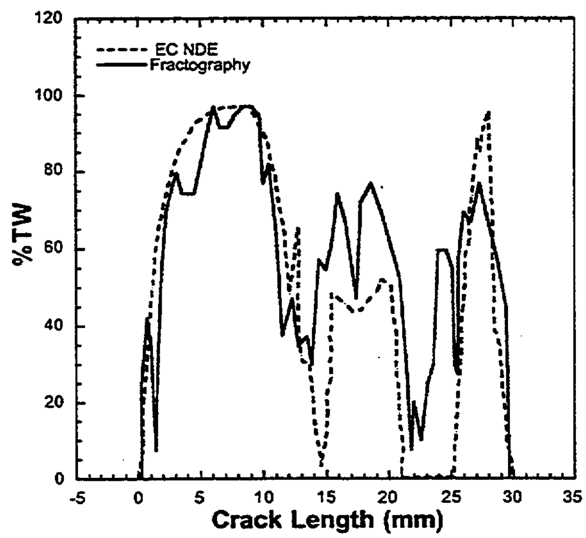


Figure 3.
Destructive analysis of 29 tube set was performed to validate the capability of multifrequency phase analysis to better size and characterize a variety of flaws. Three examples are shown.

study. In all of these studies, it is desirable to conduct the tests in the appropriate crevice chemistry environments.

Finally, studies on the behavior of Alloy 600 are still important, even though replacement steam generators are using Alloy 690 tubes. We have extensive and very valuable field experience with Alloy 600 that can be coupled with laboratory data. The knowledge gained in this coupling process can provide a bridge between laboratory data and expected field behavior for Alloy 690. The two alloys should be studied under similar conditions, and a better understanding of crack initiation, evolution, and growth under realistic crevice chemistry conditions is needed for both materials.

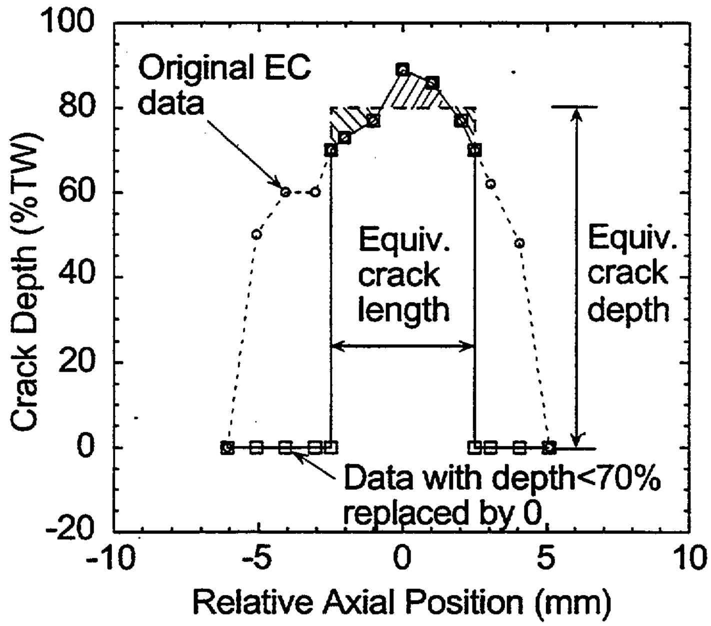


Figure 4.
"Equivalent rectangular crack" methods are promising, but more validation work on a wider variety of crack geometries is needed.

6. Observations and Recommendations

6.1 NDE

There is a need for a more robust screening of SG tubing. The development of array probes, which have better resolution than bobbin coils and enable rapid detection of both axial and circumferential cracks, may be the route to improved screening.

The industry has developed inspection technologies, performance demonstration and qualification programs that have improved the effectiveness and reliability of steam generator inspection programs, but improvements are needed in inspection guidelines and performance demonstration required to qualify techniques and analysts. The current passing criteria is 80% detection rate. Is it acceptable to have 20% of deep flaws remaining in service?

Currently, there are no passing criteria for sizing. Qualification for sizing is needed using a sample sets with realistic cracks and other flaws. Until better sizing can be achieved, degradation of steam generator tubing by stress corrosion cracking may have to be managed by "plug or repair on detection."

Replacement SGs with 690 tubing may require a different approach to ISI. In general larger samples are needed for early detection of developing degradation. Thus it may be preferable to have 100% sampling rather than the 3% or 20% often proposed currently. The larger inspection sample could be balanced by a lower frequency of inspection, particularly in early years of operation. In addition, a higher POD for smaller flaws is needed for early detection.

6.2 Integrity

Currently, reliable correlations for predicting structural integrity and leakage of tubes with single well-defined rectangular cracks or notches are available. These models can be used to conduct conservative calculations by replacing actual cracks by bounding rectangular cracks. However, these evaluations can sometimes be overly conservative. To obtain more realistic assessments, these models have to be extended and/or modified.

The "equivalent rectangular crack" appears to be a reasonable approach towards a more realistic description of planar cracks with irregular shapes.[32,33] The current approach projects the crack depth profile as measured from NDE on to a single plane and treats the crack as a planar crack. But tests show that such planar cracks tend to have lower ligament rupture and burst pressures and higher leak rates compared to cracks that are non-planar and segmented separated by ligaments. A key to developing more realistic rupture and leak rate criteria is to determine the behavior of such ligaments as a function of their size and width.

An interesting recent observation made on deep stress corrosion cracks is the significant time-dependent increase of leakage under constant pressure hold, indicating an increase of throughwall crack length due to time-dependent ligament rupture. Such behavior has been observed in tests at room temperature as well as at 282°C - a temperature regime where time-dependent creep deformation is generally accepted to be negligible. Such a time-dependent behavior under constant pressure suggests that the ligament rupture pressure of deep cracks under a constantly rising pressure test may be dependent on the pressurization rate. An analogous effect of pressurization rate on ligament rupture pressure has also been observed for deep planar machined notches with variable ligament width. Currently, we have no model to account for such time-dependent ligament rupture phenomenon.

To select appropriate integrity model used in the operational assessment, we need to better predict how flaws develop, evolve, and grow from more complex infant cracks to planar cracks. Again, the behavior of ligaments under pressure and corrosive environment is the key to developing such predictive models.

Nucleation and early growth of stress corrosion cracks are controlled by various factors – some of them are mechanical (e.g., stress), some are environmental (e.g., temperature), some are chemical (e.g., pH) and some are metallurgical (e.g., carbide morphology). Currently, the complex interactions between these various factors are not clearly understood. Also, crack morphology during this period can be very complex (e.g., cellular cracks rather than a single dominant crack). As a result, mechanistic models for predicting crack initiation are currently lacking and empirical models based on stress are often used for predicting crack initiation. With continued service exposure, often a dominant single crack emerges for which the mechanical component of the crack driving force becomes controlling. Fracture mechanics-based models can then be used to calculate the growth of such cracks and crack growth rate data for alloy 600 under primary and secondary water environments have been generated for this purpose.

6.3 Materials Degradation

As noted previously service experience to date with thermally treated Alloy 690 has been good. After up to 11 years of service, no incidents of SCC have been reported in operating steam generators. However, although laboratory studies have also been unable to produce SCC in Alloy 690 in primary water chemistries, numerous studies have demonstrated the ability to crack this alloy under conditions that approximate chemistries that could occur under crevice conditions on the secondary side of steam generators. In addition, it should be noted that widespread instances of SCC with Alloy 600 tubes did not occur until after ≈10 years of service. The situation has some similarity to that in the late '60s when Alloy 600 was thought to be the solution to steam generator corrosion problems. Although designs and water chemistry controls have improved and Alloy 690 is clearly a more resistant material, it should not be assumed that the SCC problem in PWR steam generators has been permanently solved through this choice of materials.

Studies of Alloy 600 behavior are still important, even though replacement steam generators use Alloy 690 tubes. There is extensive field experience with Alloy 600 that can be coupled with laboratory data to help understand and validate the relation between laboratory data and behavior in actual steam generators. This information can be used to provide a bridge between laboratory data and field behavior for Alloy 690.

Substantial progress is being made in improving NDE capability and developing a better understanding of the structural behavior of flawed tubes. However, we still need to gain a better understanding of crack initiation, evolution, and growth under realistic crevice chemistry conditions for both Alloy 600 and 690 to carry out more realistic operational assessments of steam generator integrity.

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Technical Issues Stemming from Recent Steam Generator Experience at Indian Point 2 and Arkansas 2

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28th Water Reactor Safety Meeting
Bethesda, Maryland
October 24, 2000

Purpose

To identify some of the issues stemming the NRC staff's review of steam generator (SG) tube integrity problems experienced at Indian Point 2 and Arkansas Nuclear One, Unit 2.

February 15, 2000 SG Tube Failure at Indian Point 2 (IP-2)

- Resulted in 150 gpm leak (primary to secondary)

- Failure mechanism:

  - primary water stress corrosion crack (PWSCC) located at apex of small radius u-bend

  - abnormal stress level due to tube ovality induced by leg displacement of the u-bends

    - » denting at the tube to tube support plate (TSP) intersections

    - » “hourglass” deformation of the upper TSP flow slots

NRC Staff Response to IP-2 Event

- Augmented Inspection Team (AIT)
    - AIT report dated April 28, 2000 (accession no. ML003709064)
- Special NRC team inspection
    - Special Inspection Report dated August 31, 2000 (accession no. ML3746339)
    - Addressed causes of IP-2 event, including licensee performance during 1997 SG inspection.
    - Identified licensee performance issues.

NRC Staff Response to IP-2 Event (Continued)

- Technical Evaluation Report (TER) dated October 11, 2000 (accession no. ML003759418)

    - Addressed the root causes of the event, post-event inspections and corrective actions, and the licensee's operational assessment to justify plant restart and continued operation.

    - Staff reached no conclusions regarding plant restart at time licensee elected to proceed with SG replacement.

    - The TER cites a number of technical issues pertaining to the licensee's operational assessment.

Performance Issues Pertaining to 1997 Inspections at IP-2

Two of the performance issues cited in the NRC special inspection report involved failure of the licensee during the 1997 SG inspection to identify and follow up on precursors for the mechanism which ultimately led to the failure event.

- The failure mechanism for IP-2 is the same that caused a similar event at Surry 2 in 1976.
- Denting activity was known to be present at the upper TSP.
- Significant hourglass deformation was known to be present in lower TSPs.
- Licensee did not have adequate procedure or criteria for identifying significant hourglass deformation in upper TSP.
- Small radius u-bend indication found by inspection. Tube integrity implications of this indication was not adequately assessed.

Performance Issues Pertaining to 1997 Inspections at IP-2

Poor quality of the eddy current test (ECT) data was a third performance issue cited in the NRC special inspection report.

- Look-back (hindsight) analyses of 1997 ECT data revealed presence of 4 additional PWSCC indications at the apex of the small radius u-bends missed during the 1977 data analysis.
    - mid-range plus-point probe, 300 KHz
    - the missed indications were masked by noise stemming from surface deposits and mechanical sources
- The licensee failed to recognize the noise as a "condition adverse to quality."
- The licensee did not take steps to improve data quality or establish data quality acceptance criteria.

Operational Assessment Issues - IP-2

Background

- Operational assessments are performed to demonstrate that adequate structural margins and leakage integrity will be maintained until the next scheduled inspection.
    - Considers beginning of cycle (BOC) flaw distribution based upon inspection results and assumed NDE flaw detection and sizing accuracy capability.
    - Flaw growth rates are applied to BOC distribution to yield projected end of cycle (EOC) distribution.
    - Structural margins and leakage integrity are evaluated for projected EOC flaw distribution viz-a-viz applicable limits.

Operational Assessment Issues - IP-2

- Staff had reached no conclusions regarding plant restart at time licensee elected to proceed with SG replacement.

- The licensee was unable to resolve staff concerns relating to uncertainties in the NDE flaw detection and sizing performance parameters assumed in the operational assessment. Additional concerns existed with respect to the operational assessment methodology.

Operational Assessment Issues - IP-2

- The post-event, 2000 inspection of the small radius u-bends was performed with the high frequency plus-point probe (@800 KHz) for flaw detection and the mid-range plus-point probe (@ 400 KHz) for flaw sizing.

- The industry qualification data sets for NDE flaw detection and sizing performance in small radius u-bends consists largely of tube specimens containing EDM notches rather than real cracks.

  - EDM notches are easier to detect and size than real cracks.

- A qualification data set was available for the mid-range plus-point probe (@ 300 KHz) for the case of actual PWSCC flaws at dented egg crate support plates.

  - Performance data from this qualification set was assumed to apply to the u-bend inspections at IP-2.

Operational Assessment Issues - IP-2

- Licensee was unable to demonstrate to staff's satisfaction that the tubing essential variables of the qualification data set were representative of those of the small radius u-bends at IP-2.
    - A major consideration for the staff was the comparative noise levels between the qualification data set and the IP-2 u-bends.

In-situ Pressure Test Failures at ANO-2

Background: Condition monitoring (which may include in-situ pressure testing) is performed during each inspection outage to confirm that all tubes maintained adequate structural margins throughout the previous operating cycle.

- During a refueling outage SG inspection in January 1999, a tube exhibiting outer diameter stress corrosion cracking (ODSCC) near an egg crate TSP failed to satisfy the applicable structural criterion (3 delta P) during an in-situ pressure test.

- A mid-cycle inspection was conducted in November 1999. In-situ testing was terminated for one tube due to leakage in excess of system capacity before reaching a test pressure equivalent to the 3 delta P criterion.

ANO-2 Issues - Tube Selection Criteria for In-Situ Testing

- Selection criteria addressed in industry guidelines
  - Should account for NDE flaw size measurement error
- In November 1999, the licensee identified six tubes as meeting the selection criteria for performing the test.
  - The licensee initially determined that 4 of these tubes need not be tested since the measured flaw size was bounded by measured flaw sizes for tubes which previously in-situ pressure tested successfully.
- The staff stated, during discussions with the licensee, that these four tubes should be tested to account for NDE measurement error.
- In response, the licensee tested these 4 tubes, one of which was the tube for which the test was terminated prior to reaching 3 delta P.

ANO-2 Issues - Interpretation of In-Situ Test Results

- The licensee assessed the circumstances of the test results and concluded that the leakage corresponded to ligament tearing of the flaw and that the tube burst pressure exceeded the 3 delta P crit.
  - With this as a benchmark for its operational assessment, the licensee concluded it could operate to its scheduled refueling outage in September 2000 while maintaining adequate margin.

- The staff reviewed the basis for these findings and disagreed with both of these conclusions, based on consideration of the results of the test and staff calculated burst pressure associated with the pretest NDE flaw profile.

- The staff's findings were documented in:
  - NRC letter dated May 2, 2000 (accession no. ML003710343)
  - NRC letter dated June 23, 2000 (accession no. ML003726321)

ANO-2 Issues - Pressurization Rate Issue

- The licensee pressure tested a number of lab tube samples with EDM notches to provide additional evidence in support of its previously stated conclusions.

- These tests failed to provide this evidence.

- Further, these tests revealed that the burst pressure of these lab samples was strongly affected by the pressurization rate employed during the tests.

    - unexpected finding, based on prior industry data

- Industry is investigating the causes of this apparent pressurization effect and the potential generic implications for pressure test procedures and existing burst pressure data bases.

ANO-2 Issues - Pressurization Rate Issue

- Preliminary findings from the industry are:
  - The pressurization rate effect is limited to planar, part thru-wall flaws with maximum depths greater than 90% thru-wall.
  - Existing burst pressure data bases are not significantly affected.
  - Changes to industry pressure test procedures are needed to ensure burst pressure data not influenced by pressurization rate.
- A more description of this issue and the status of the industry investigation is described in the following NRC meeting summaries:
  - NRC meeting summary dated July 6, 2000 (accession numbers ML003761447 and ML003761349)
  - NRC meeting summary dated September 28, 2000 (accession no. ML003760794)

ANO-2 Issues - Bench marking of Operational Assessments

- The licensee submitted a request to change the plant licensing basis (3 delta P) to allow operation until the September 2000 refueling outage on basis of risk informed demonstration that SG tube integrity would meet the acceptance criteria in NRC Regulatory Guide 1.174 which deals with the use of PRA in risk informed decision making.

- The staff rejected this request because of uncertainties in the tube integrity margins that would exist by September 2000.

  - The licensee was unable to benchmark to the staff's satisfaction its supporting operational assessment methodology to previous inspection results during the February 1999 refueling outage and November 1999 mid-cycle inspection.

  - This called into question the assumed NDE flaw detection performance and flaw growth rates used in the analysis.

ANO-2 Issues - Bench marking of Operational Assessments

- A more detailed description of the staff's evaluation is documented in an NRC safety evaluation dated July 21, 2000 (accession no. ML003734450).

EPRI MATERIALS RELIABILITY PROJECT: MASTER CURVE ACTIVITIES

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ABSTRACT

Recent nuclear industry attention has been focused on the direct use of measured fracture toughness properties in the assessment of RPV integrity. Specifically, efforts have been initiated to develop procedures for determining the material transition temperature based on measured fracture toughness testing (T_0) using the Master Curve approach. The direct determination of material fracture toughness, and a transition temperature associated with measured fracture toughness, represents a more precise measure of material resistance to crack initiation than earlier methods and should provide a more realistic assessment of RPV integrity. The EPRI Materials Reliability Project (MRP), through the Reactor Pressure Vessel Integrity Issue Task Group (RPV Integrity ITG) is an active participant in the coordination of U.S. industry activities supporting the application of the Master Curve approach for RPV integrity assessment. The primary goal of the Master Curve program is to resolve the various technical issues associated with application of the Master Curve approach for use in RPV integrity assessment. The Master Curve approach is a demonstrated technically superior method to assess RPV material condition. The objective of the MRP activity is to validate implementation of modern fracture toughness testing technology to permit direct measurement of vessel embrittlement. More specifically, the goal is to facilitate characterization of lower bound toughness in the transition region of irradiated steel using precracked Charpy specimens. The program consists of: proof of principal analysis; development of a physical basis for the master curve; facilitation of the development of testing and implementing standards and codes; development of plant specific submittal strategies; resolution of technical issues; evaluation of margins; and evaluation of the effect of use of the Master Curve on vessel risk. This paper provides a brief overview of the EPRI MRP program, the RPV Integrity ITG, and industry activities associated with development of the Master Curve approach for RPV integrity assessment. Specifically, this paper will provide details on the development of a physical basis for the Master Curve and margins evaluations being performed to support application of the Master Curve.

INTRODUCTION

The direct measurement of reactor pressure vessel (RPV) fracture toughness is an important element in the continued demonstration of RPV integrity. Recent nuclear industry attention has been focused on the direct use of measured fracture toughness properties in the assessment of RPV integrity. Specifically, efforts have been initiated to develop procedures for determining the material transition temperature based on measured fracture toughness testing (T_0) using the Master Curve approach. This is in contrast to the historically utilized approach to first determine an initial unirradiated reference temperature, RT_{NDT}, based on a combination of Charpy V-notch and drop weight nil-ductility transition (NDT) temperature test methods, and adjust for neutron radiation damage by adding the Charpy transition temperature shift measured at 30 ft-lb. The direct determination of material fracture toughness, and a transition temperature associated with measured fracture toughness, represents a more precise measure of material resistance to crack initiation than earlier methods and should provide a more realistic assessment of RPV integrity.

Significant efforts are presently underway in the research, codes and standards, and regulatory arenas to:

- Collect all relevant and available fracture toughness data for RPV materials
- Complete industry test programs on non-irradiated and irradiated RPV materials (e.g., Owners Groups, IAEA Coordinated Research Program, PVRC/MPC round robin, lead plant testing of irradiated surveillance materials, and testing of selected irradiated materials funded through EPRI)
- Develop consensus procedures for the determination of T_0
- Demonstrate the validity of using T_0 as a fracture toughness transition reference temperature
- Incorporate the new approach into appropriate codes, standards, and regulations

These activities involve the participation of all aspects of the nuclear industry including individual utilities, nuclear steam supply system (NSSS) vendors, vendor Owners Groups, EPRI, research laboratories, the U.S. Nuclear Regulatory Commission (NRC), and other domestic and international experts. Many of the industry activities in this area are being coordinated through the EPRI Materials Reliability Program (MRP). To more fully appreciate the emphasis that the industry has placed on pursuing the Master Curve approach for RPV integrity assessment it is necessary to provide a brief background on the EPRI MRP.

EPRI MATERIALS RELIABILITY PROGRAM (MRP)

The MRP was formed in 1998 and chartered to implement and maintain an industry wide program focused on resolving selected existing and emerging pressurized water reactor (PWR) materials performance, safety, reliability, operational and regulatory issues. The MRP provides a single utility-directed oversight structure to proactively address and resolve, on a consistent industry-wide basis, selected PWR material-related issues. The Executive Group of the PWR MRP now serves as the industry focal point for resolution of issues related to PWR materials degradation management. Key elements are summarized below.

The specific objectives of the MRP are:

- To resolve existing and emerging PWR materials performance, safety, reliability, operational and regulatory issues that meet specific screening criteria;
- To serve with the direct involvement of the Nuclear Energy Institute (NEI) as the focal point for industry-wide PWR materials-related regulatory issues; and
- To fully integrate any work undertaken with Owners Group activities and, where appropriate, American Society of Mechanical Engineers (ASME) Code activities on these issues.

The MRP executive oversight committee, the PWR Materials Management Program (PMMP) Steering Committee, consists of at least ten executive line managers. Each NSSS Owners Group is represented on the PMMP Steering Committee. NEI and Institute of Nuclear Power Operations (INPO) representatives will be included in the PMMP Steering Committee in a liaison role. The PMMP Steering Committee will formulate recommendations for overall program direction, strategy and funding, and will approve issues to be addressed, budget plans and final issuance of issue resolution documents. The PMMP Steering Committee will also be the principal interface, through NEI as appropriate, with the NRC for issues having a regulatory component.

The Senior Representatives, acting as a committee, will: 1) provide the authority for the PMMP Steering Committee to act as an executive committee; 2) provide general policy direction; 3) approve PMMP Steering Committee recommendations for supplemental funding support as needed; 4) make utility commitments for their respective utilities; and 5) provide for interaction with the NRC. The Senior Representatives provide final authority for overall funding.

An Integration and Implementation Group (IIG) was established to recommend to the PMMP Steering Committee the issue(s) to be addressed, the needed resources to address the issue(s) and the makeup of the Issue Task Groups (ITGs) to address the issue(s). Additionally, the IIG will concur in the ITG-developed priorities and approach to reaching issue closure. NEI and INPO representatives will be included in the IIG in a liaison role. When appropriate, NSSS vendors may be invited to participate in the IIG meetings in an information exchange role.

The MRP organization below the IIG level includes four Issue Task Groups (ITGs) and a Technical Support Subcommittee (TSS) reporting to the IIG. NEI and the Owners Groups are represented on the IIG and on each ITG. The PWR NSSS vendors are also represented on the ITGs. The Technical Support Subcommittee (TSS) has been formed to address materials issues and technology development needs that fall outside of the specific focus of the three ITGs. The TSS addresses generic PWR/BWR materials and inspection issues that require coordination with other EPRI programs and involvement of BWR as well as PWR utility representatives. Figure 1 shows the organizational structure of the MRP.

All MRP communications with NRC are coordinated through NEI. For some issues, individual Owners Groups communicate directly with NRC, in which case overall responsibility for coordination and consistency of communications resides with IIG and NEI.

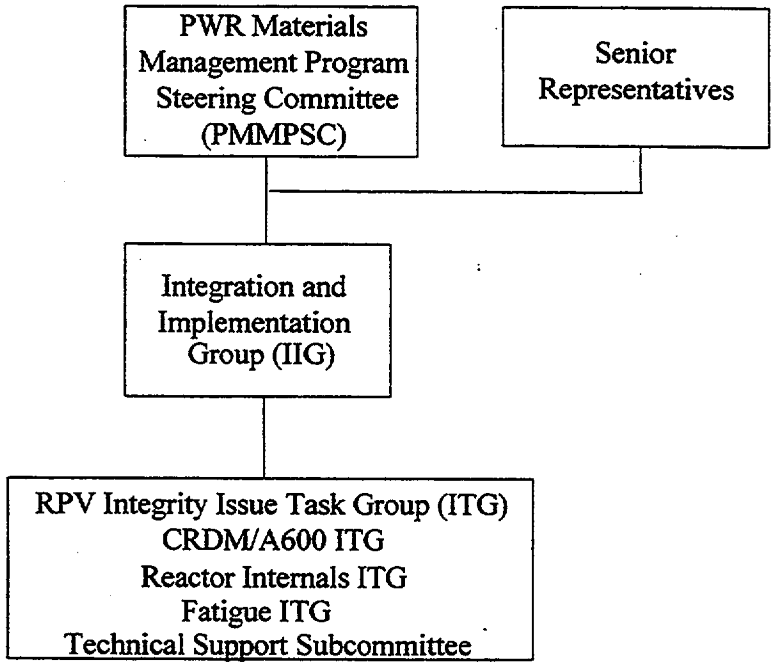

Figure 1-1
EPRI PWR MRP Organization

Issue Selection Criteria

An early action of IIG was to establish criteria for selection of issues requiring a unified approach to resolution and executive-level oversight. The MRP undertakes resolution of issues that satisfy all or many of these criteria:

1. Resolution is long-term, probably requiring two years or more.
2. The issue has a potentially large financial impact due to plant shutdowns or major component repair or replacement.
3. The issue has regulatory implications and potential safety impact, either at the outset or in the future.
4. The issue currently or potentially impacts a large number of plants.
5. The issue generally impacts plants of more than one NSSS vendor.
6. Generally impacts international (non-US) utilities and the information and participation of such organizations is desirable.
7. Typically the solution to the issue will involve a focused effort in "aging management" which consists of a mix of needs/options dealing with application (and possible development) of inspection capability and technology, repair methods and technology, mitigation measures and safety/operability analysis methods.
8. Consensus is difficult to achieve, due to diverse technical or institutional perspectives.
9. MRP resolution is requested by or agreed to by the Owners Groups.
10. Resolution is most efficient if done centrally.

MRP Issue Task Groups

During early 1998, the IIG reviewed virtually all of the materials-related work planned or in progress by EPRI and the Owners Groups. Based on the selection criteria above, four issues were recommended and eventually approved for MRP action and resolution. Issue Task Groups (ITGs) were formed for each of these: (1) RPV Integrity ITG, (2) Reactor Internals ITG, (3) Control Rod Drive Mechanism (CRDM)/Alloy (A) 600 ITG, and (4) Fatigue ITG. The RPV Integrity ITG is discussed next.

RPV INTEGRITY ISSUE TASK GROUP

The primary objective of the RPV Integrity ITG is to eliminate reactor vessel integrity challenges to plant operation that could lead to premature shutdown. The activities under way in the ITG program are intended to lead to the following results:

1. Open operating windows for plant heatup and cooldown
2. Ensure that all plants remain below the PTS screening criteria through end-of-life (EOL) and if possible through a license renewal period
3. Provide new technologies to more accurately determine RPV embrittlement
   - Master Curve approach for direct measurement of embrittlement
   - New ASTM E-900 embrittlement correlation
4. Revision to NRC regulations to allow use of new technologies

The RPV Integrity ITG efforts are focused in the following areas: (1) reevaluation of the pressurized thermal shock (PTS) screening criteria; (2) validation of the Master Curve fracture toughness approach for vessel integrity assessment; (3) development of an updated embrittlement correlation through ASTM Standard Guide E900; and (4) other technical areas such as development of material property databases. The Master Curve program is described below.

MASTER CURVE PROGRAM

The primary goal of the Master Curve program is to resolve the various technical issues associated with application of the Master Curve approach for use in RPV integrity assessment. The Master Curve approach is a demonstrated technically superior method to assess RPV material condition. The objective of the MRP activity is to validate implementation of modern fracture toughness testing technology to permit direct measurement of vessel embrittlement. More specifically, the goal is to facilitate characterization of lower bound toughness in the transition region of irradiated steel using precracked Charpy specimens.

Successful demonstration of the Master Curve approach and resolution of technical/application issues will directly benefit light water reactor owners through a better characterization of RPV integrity. This will be manifested through additional operating flexibility during reactor startup and shutdown for both PWRs and boiling water reactors (BWRs) and additional margin against present (or revised) PTS screening criteria.

The MRP is actively pursuing validation of the Master Curve fracture toughness approach for application to RPV integrity. The MRP supported development of ASME Section XI and Section III Code Cases allowing use of the Master Curve in developing a fracture toughness-based reference temperature, RT_{T_0} [1,2]. The MRP developed and published the technical basis document [3]. The RPV Integrity ITG also provided technical review of a plant-specific licensing action by Wisconsin Public Service regarding use of the Master Curve.

The MRP RPV Integrity ITG Master Curve program consists of the following activities:

Proof of Principle Analyses

This task area involves empirical validation of Master Curve applicability to RPV materials through the collection and evaluation of all relevant Master Curve data. The comprehensive database established by the MRP for this task is also being used to support the resolution of technical issues identified regarding application of the Master Curve approach for RPV integrity assessment. Additional details regarding the data collected and evaluated under this task has been described elsewhere [4].

Resolve Technical Issues Associated with Application to RPV Integrity Assessment

In applying the Master Curve method to the pressure vessel industry, there are a number of technical issues to be clarified among the users and the regulators. The key technical issues and industry efforts to resolve them are summarized below. Detailed technical discussions of these issues are provided in Reference [4].

Validity of three-point bend tests using pre-cracked Charpy specimens from surveillance programs

Reactor surveillance programs in general do not include a sufficient number of fracture toughness specimens to allow a determination of T_0. This difficulty may be overcome by allowing the reactor vessel owner to test Charpy-sized specimens from reactor surveillance programs as pre-cracked three-point bend specimens using the Master Curve method. In principle, there is nothing to preclude this option as long as use of the Charpy specimen does not invalidate any of the basic premises underlying the Master Curve methodology. It is quite possible that both in-plane constraint loss and inaccuracies in the thickness correction for these small specimens may produce a non-conservative bias in the T_0 measurement. A number of tests performed by different laboratories and the U.S. PWR Owners Groups confirm the validity of three-point bend tests using pre-cracked Charpy specimens. Application to Charpy-size specimens requires judicious selection of the appropriate test temperature, and more than six test specimens may be required to obtain a valid measurement of T_0. An additional round-robin study (International Atomic Energy Agency Coordinated Research Program, IAEA CRP) using pre-cracked Charpy specimens has provided further verification of the use of this type of specimen for the Master Curve [5]. Additionally, members of a Pressure Vessel Research Council (PVRC) Task Group will be completing a series of tests to establish the validity of the Weibull statistics for the pre-cracked Charpy specimen tests.

Statistical Size Scaling for RPV Applications

For cleavage initiation toughness there is a clear statistical relationship between the measured fracture toughness and the length of the crack front. While Master Curve technology provides procedures to accommodate this statistical size effect, current regulations do not include this effect. This difference between the two approaches can produce apparent inconsistencies in the application strategy. This issue may be addressed by carefully defining the crack size assumptions included in the analysis. Efforts by the NRC and the EPRI MRP under the PTS Reevaluation Effort are currently evaluating more realistic flaw size distributions for reactor pressure vessel applications. These flaw size distributions should also be relevant to Master Curve analysis.

Use of the alternate reference temperature approach to account for dynamic effects

The use of the Master Curve method to establish an alternate reference temperature index assures that measured values of the static fracture toughness do not exceed the ASME Code K_{IC} reference toughness curve. However, this reference temperature is also used as an index to the ASME Code K_{IR} reference toughness curve, which bounds dynamic and crack arrest data. Use of the alternate reference temperature approach relies on the relationship between the K_{IC} and K_{IR} curves to properly account for the dynamic effects on fracture toughness. Recent dynamic tests conducted suggest that dynamic fracture toughness data do conform to the RT_{T_0} indexing method for a mild steel (A515); three dynamic loading rates were utilized in that study [6].

A linear relationship has previously been established between T_0 and the logarithm of stress intensity factor rate, dK_I/dt [6] A similar linear trend has been established for ferritic reactor pressure vessel materials over this range of loading rates [7]. A previous model relating the dynamic fracture toughness to the material yield strength works reasonably well for this material [8]. For nuclear power plant applications, the dynamic loading rate is usually small. The most severe PTS (pressurized thermal shock) transients have moderately low initial dynamic loading rates, dK_I/dt of less than 1.3 ksi$\sqrt{\text{in}}$/s. However, ductile crack initiation and arrest during PTS can result in re-initiation at higher loading rates. From these data, a shift for static versus dynamic data is not expected to be greater than approximately 30°C for low to moderate loading rates. Additionally, analysis of a PVRC database [9] indicates that higher loading rates also do not shift the fracture toughness data more than 30°C relative to static data. This shift is consistent with the difference between the ASME Code K_{IC} and K_{IR} reference toughness curves. While arrest is mechanistically different than crack initiation, an empirical relationship has been demonstrated of T_0 as an index temperature for a crack arrest toughness curve [10].

The basis assumed for the ASME Code Cases N-629 and N-631 is the fixed difference in temperature space between static and dynamic/arrest behavior. This difference is generally conservative for unirradiated materials and becomes even less for irradiated materials. Thus, an added conservatism is included when applying Code Case N-629 for irradiated material condition. This issue becomes important when future efforts are pursued to replace the existing ASME Code fracture toughness curves with a Master Curve tolerance bound based upon static fracture toughness measurements.

Irradiation may alter the shape of the transition curve

The consistency of transition curve shape between different materials is a fundamental assumption of the Master Curve approach. Given the wide range of ferritic steels that have been successfully analyzed using the Master Curve approach, it is reasonable to assume that irradiated reactor pressure vessel steels will behave similarly. However, there is some concern that irradiation might alter the shape of the transition curve. The concern over the shape of the Master Curve might arise by analogy with Charpy behavior. It is true that Charpy curves have different transition region shapes and upper shelf levels, depending upon the degree of radiation embrittlement and the Charpy upper shelf energy level. However, fracture toughness tests in the transition range have shown that the fracture toughness shape does not change with irradiation. A detailed analysis of the physical basis for the shape consistency of the Master Curve is being conducted by the MRP and will be discussed later.

Material variability must be accounted for when transition temperature measurements from surveillance materials are applied to RPV analysis

The surveillance materials are specifically selected to be representative of the limiting materials used in the fabrication of the reactor pressure vessel. The use of representative materials to define the properties used in integrity analysis is standard engineering practice. Conservatism can be added to the analysis through the use of appropriate margins. Recent questions about the degree of variability in the radiation response of several critical reactor pressure vessel materials has generated concern about the use of properties determined from surveillance materials in the analysis of nuclear pressure vessel steels. The development of an appropriate margin strategy requires precise definitions of the material variability issues involved. Activities in this area will be discussed later.

Facilitate Development of Testing and Implementation Standard and Codes

For benefit to be realized by the nuclear industry regarding application of the Master Curve approach, relevant codes and standards that govern plant operating criteria, and ultimately NRC regulatory documents, must be revised to incorporate results of research sponsored by MRP, the NRC, and the industry in general. The MRP is supporting these efforts. Recently, the MRP supported development of ASME Section XI and Section III Code Cases allowing use of the Master Curve in developing a fracture toughness-based reference temperature, RT_{To} [1,2]. The MRP developed and published the technical basis document [3]. The MRP will also pursue additional changes in appropriate operating plant criteria as research results warrant.

Development of Plant-Specific Submittal Strategies

The MRP has provided technical support, in terms of strategy development, independent technical review, and other activities, to MRP members that are pursuing licensee submittals that utilize the Master Curve technology. Recently, the RPV Integrity ITG provided technical review of a plant-specific licensing action by Wisconsin Public Service regarding use of the Master Curve. This support will continue.

Evaluate the Effect of Using the MC on Overall Vessel Risk

Efforts are presently underway between the EPRI MRP and the NRC to develop a technical basis for revising the PTS Screening Criteria contained in 10CFR50.61. Advances in probabilistic risk assessment (PRA), probabilistic fracture mechanics (PFM), thermal hydraulics (TH), and overall plant risk assessment are being incorporated into a comprehensive program to establish a technical basis for revising the present screening criteria. The MRP plans to perform various PFM sensitivity studies to investigate the overall impact of various input parameters on overall vessel risk. Included in this analysis will be an assessment of the impact of using the Master Curve approach on the overall vessel failure risk under postulated PTS transients. Consideration of appropriate margins and uncertainty will be a key aspect of this sensitivity study.

Development of a physical basis

The Master curve approach to characterizing fracture toughness transition behavior of pressure vessel steels is based on statistical analysis of empirical data and not on a physics-based understanding of the fracture behavior of these steels. While there is an abundance of empirical data that supports the idea of a single curve shape for all pressure vessel steels this cannot replace the need for solid physical modeling. The lack of directly measured fracture toughness data for a considerable proportion of the nuclear fleet suggests that a purely empirical argument cannot validate the Master Curve for all conditions of interest. In order to validate the Master Curve approach research has been undertaken to provide a physical understanding of the fracture behavior of pressure vessel steels. This work provides the basis for defining the limits of applicability of the Master Curve as well as for enabling extrapolation of the Master Curve model to other material conditions with only limited testing. A model has been developed which is based on the physical mechanisms of transition fracture and combines equations of material plasticity with continuum models of crack tip stress fields to predict fracture toughness transition behavior with temperature of ferritic steels.

The basis for derivation of a physically-based model is the Orowan-modified Griffith equation in which the effective energy to fracture in the fracture transition region is assumed to be dominated by the plastic work [11]. This is incorporated into the Griffith equation via a plastic work term to modify the energy absorbed in fracturing a material. Wallin et al. [12] suggested that the temperature dependence of the plastic work controls the observed exponential temperature dependence of the fracture toughness. They further state that the empirically-derived, exponential temperature dependence of w_p is based on the Peierls-Nabarro stress that describes the resistance of the lattice to dislocation motion.

Natishan and Kirk [13] used this as the basis of their proposed, physically-based model for plastic work to fracture in which they suggest that w_p can be defined as the strain energy density (area under the true stress-strain curve) taken over some microstructurally significant length scale.

$$w_p = \int_0^{\varepsilon_c} \sigma d\varepsilon \cdot \ell \tag{1}$$

Choosing an appropriate constitutive flow curve equation was the key to understanding the physical basis for the observed temperature-dependence of w_p. They chose the Zerilli-Armstrong (ZA) [14] equation:

$$\sigma_{ZA} = c_0 + B_0 e^{-\beta T} + K\varepsilon^n \tag{2}$$

where

$$\beta = \beta_0 - \beta_1 \ln \dot{\varepsilon} \tag{3}$$

and

$$c_0 = \sigma_G + kd^{-1/2} \tag{4}$$

Strain is denoted by ε, $\dot{\varepsilon}$ is the strain rate, T is the absolute temperature, d is the average grain diameter and σ_G is the contribution to the flow stress from solutes and the initial dislocation density. The values k, β_0, β_1, B_0, K and n are constants specific to a material.

Wagenhofer et al. [15] developed this idea further following the work of Tetelman, Wilshaw and Rau [16 and 17], and, more recently, Chen and Wang [18] in which three criteria for fracture of mild steel in the transition region were outlined. These criteria are:

1. Microcrack nucleation -- this has been well documented to be the result of dislocation motion and accumulation at long range obstacles such as grain boundaries, and second phase particles;
2. Propagation of the microcrack through the grain in which it was nucleated; and
3. Propagation of the microcrack through boundaries surrounding the nucleating grain.

Wagenhofer et al. [15] incorporate the critical strain to fracture a carbide as the limit of integration in the equation for the effective plastic work to account for microcrack nucleation. This critical strain is determined by solving for strain in the ZA equation using the carbide fracture stress.

Wagenhofer et al. [15] modified the equation for plastic work with a stress triaxiality term before the integrand to describe the competition between the ability of the stress-state to propagate the microcrack and the material's resistance to dislocation motion. This triaxiality term is thus the mean hydrostatic stress divided by the friction stress. This term ensures that the second critical event is accounted for.

The strain and triaxiality vary with distance ahead of the crack tip with the strain decreasing with x (as distance increases from the crack tip) and triaxiality increasing to a maximum at some distance away prior to decreasing. Chen and Wang [18] proposed that fracture occurred at the region of maximum triaxiality as long as the critical strain for fracture of the weakest link feature was achieved. Final fracture (the third event) occurs when the stress in the material exceeds the fracture stress.

Putting these critical events together leads to an equation for the effective strain energy to fracture:

$$\gamma_{eff} = \left(\frac{\sigma_m}{\sigma_i}\right)_f \int_0^{\varepsilon_c} \sigma_{ZA} d\bar{\varepsilon} \cdot \ell \tag{5}$$

ℓ is taken as the carbide diameter following the work of Tetelman et al. [17], suggesting that the work expended in fracturing a specimen is determined by the size of the "gage length" of material at the notch root. Tetelman et al. suggested that the region of material directly in front of the notch of a notched specimen is analogous to a series of tensile 'specimens.' The gage length of the 'specimen' is twice the notch root radius and roughly corresponds to the height of the high strain region ahead of the crack tip in which the cleavage fracture processes are likely to take place. In this case we consider growth of a carbide crack and thus it is appropriate to take the carbide radius/diameter as our 'gage length' of interest.

The temperature dependence of the plastic work term is contained within the ZA equation used to describe material plastic flow behavior and thus is used to define the critical strain required to crack the weakest link microstructural feature. Using the finite element analysis results of Chen and Wang [18] for a C-Mn steel to determine critical strain and triaxiality, and plugging these values into the proposed model for the plastic work term results in four data points (at different temperatures) that fall on the curve shown in Figure 2 representing the temperature dependence for w_p observed by Wallin et al. [12]. This understanding of the physical basis for a single curve shape for plastic work, and thus fracture toughness with temperature, should be applicable for all body-centered-cubic steels. The final results of this model development work funded through the EPRI MRP will be completed in 2001.

Assessment of Appropriate Margins for T_{oEOL} Determined Using the Master Curve

In looking to apply the Master Curve methodology to assessing end-of-license (EOL) fracture toughness transition temperature of RPV steels the opportunity arises to reassess (and redefine) the method by which margins are considered. To do this requires first establishing a clear, well defined, well understood method for obtaining a fracture mechanics-based, ductile-to-brittle transition reference temperature from whatever input data is available and then using the sensitivity studies that would have to be conducted on such a model to define margins that clearly account for uncertainties, both in the model, and due to material variability. Use of fracture mechanics advances represented in the Master Curve methodology without also redefining margins based on this methodology could decrease the benefits gained from Master Curve application.

The MRP is pursuing an approach for both establishing a method for Master Curve application in determining a ductile-to-brittle transition reference temperature (T_0) for PTS evaluation, and defining appropriate margins, based on a combination of cause-effect diagramming and PRA modeling methodologies. These methodologies are currently being used to understand and quantify the uncertainty in the RT_{NDT}/K_{IC}, PTS evaluation method. This same, well-established, approach can be used to identify causes and quantify uncertainty in proposed methods for obtaining T_{0EOL}. The uncertainty can then be used to establish appropriate margins that are consistent with application of advanced, fracture mechanics methods.

As part of the ongoing, joint industry/NRC-funded effort in PTS re-evaluation, a probabilistic risk assessment methodology has been developed to assess uncertainty in the outcome value, K_{IC}. It is based on the distribution (or uncertainty) in the input value and the model uncertainty along the path used to obtain the desired output value. This method has been demonstrated able to account for both model and input parameter uncertainty in predicting the distribution, or uncertainty, in the outcome value. This same method can be used to define appropriate margins on T_0 that account for both material uncertainty and model uncertainty.

Margins must be defined based on the methodology used to determine a reference temperature value. Thus, they can only be defined after an acceptable method for obtaining a reference temperature has been established. A program to develop a methodology for T_0 implementation in determining an EOL reference temperature has been initiated by the EPRI MRP in which the issue of margins will be addressed. A model of possible paths for obtaining T_{0EOL} is being developed (a draft is shown in Figure 3) using the cause-effect diagram method in a manner similar to that used to characterize the current RT_{NDT}/K_{IC} method of PTS evaluation. Once developed this model will be refined, using a PRA approach, into a mathematical description of the methods such that it can be used in sensitivity studies of the effects of level of material knowledge on outcome variability. Levels at which material variability can be differentiated must be defined and then distributions of input data representing the various levels of material knowledge (i.e. material class, heat, specific plate or weld, etc.) can be put into the model and their effects on the distribution of the outcome (T_0) assessed. Based on the uncertainty (or distribution) in the outcome a margin can be defined that accounts for only the uncertainty expected along a given path with a given set of input variation. Using this methodology, margins appropriate to the particular level of material knowledge and accounting for model uncertainty can be defined. The overall margins model approach (draft) is shown in Figure 4. Efforts to further refine and incorporate this model into RPV integrity assessment approaches will continue under the MRP.

CONCLUSIONS

Through the EPRI MRP, resolution of issues associated with application of the Master Curve approach for RPV integrity assessment are being addressed. Future MRP activities will focus on development of an implementation framework for RPV integrity assessment and implementation of the Master Curve methodology into plant operating criteria. Statistically defined lower tolerance bounds for unirradiated and irradiated materials can be developed directly and eventually replace the existing ASME Code reference toughness curves. Coordination of these activities with the NRC will continue.

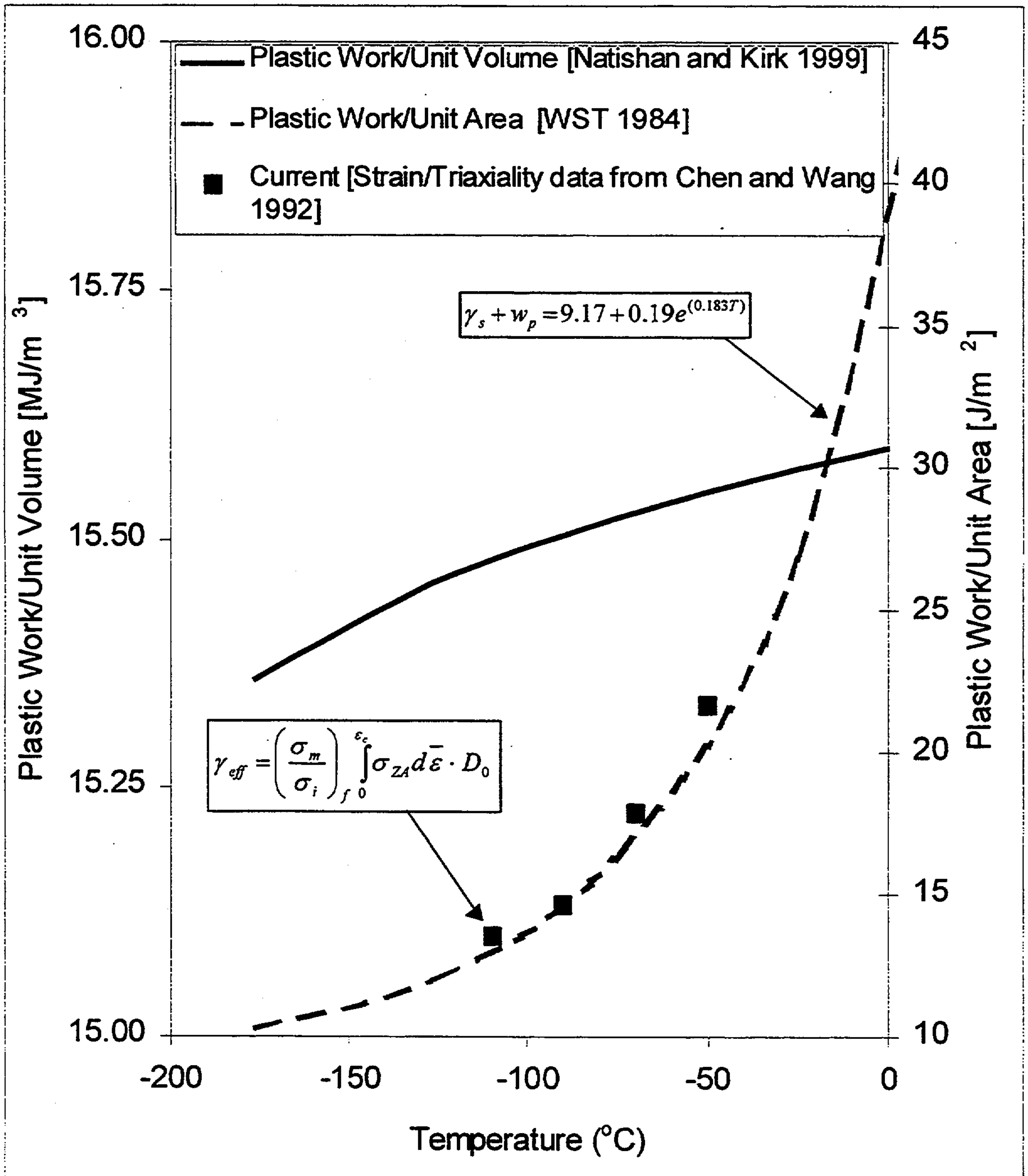

Figure 2
Comparison of Plastic Work Data versus Temperature with Prediction of Proposed Master Curve Physical Basis Model.

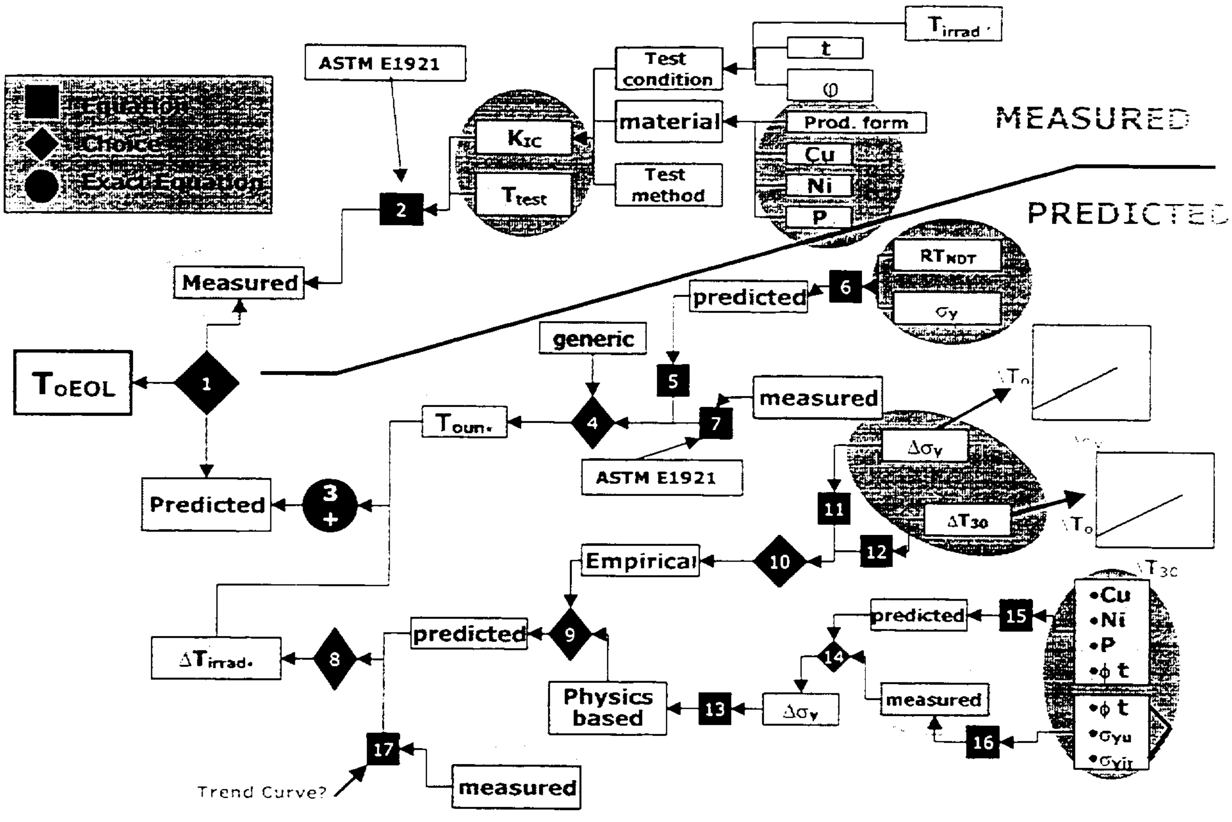

Figure 3
Draft Procedure to obtain T_{0EOL}

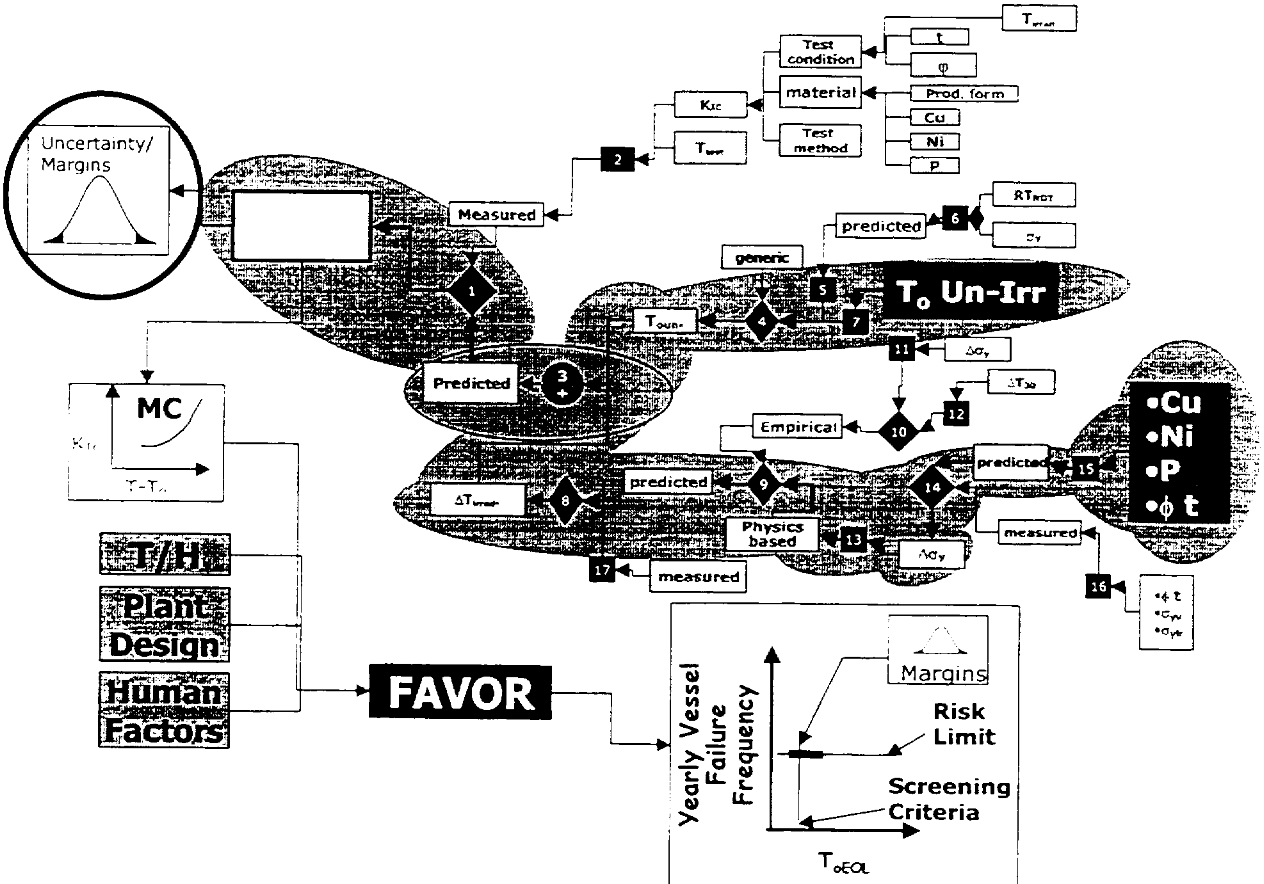

Figure 4
Draft Procedure for T_0 Margin Assessment

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NRC Review of the Technical Basis for Use of the Master Curve in Evaluation of Reactor Pressure Vessel Integrity

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BACKGROUND

Fracture Toughness Characterization

The fracture toughness of the reactor pressure vessel (RPV) steel in a nuclear plant provides a key input to calculations that commercial licensees perform to demonstrate the fracture integrity of the vessel during both normal operations and postulated accident conditions (e.g. pressurized thermal shock, or PTS). Currently, the ASME K_{IC} and K_{IR} curves, indexed to the RT_{NDT} of the material, describe the fracture toughness of the RPV and its variance with temperature. These curves were adopted in 1972 as a lower bound representation to a set of 173 linear elastic fracture toughness (K_{IC}) values and 50 linear elastic arrest toughness (K_{IA}) values for 11 heats of RPV steel. The use of RT_{NDT} to normalize temperature was intended to account for the heat-to-heat differences in fracture toughness transition temperature, thereby collapsing the fracture toughness data onto a single curve. However, RT_{NDT} is not always successful in this regard, often providing a conservative characterization of fracture toughness.

Developments since 1972 set the scene for substantial improvements to the K_{IC} / RT_{NDT} characterization of fracture toughness. In 1980 Landes and Schaffer noticed a weakest link size effect for specimens failing by transgranular cleavage. They demonstrated that larger specimens fail at lower toughness values, even when the severe size requirements of linear elastic fracture mechanics (LEFM) are satisfied. Beginning in 1984, Wallin and co-workers from VTT in Finland combined this weakest link size effect with micro-mechanical models of cleavage fracture. Wallin developed a model that accounts successfully for size effects, and provides a means to calculate statistical confidence bounds on cleavage fracture toughness data. These concepts, combined with the observation that ferritic steels exhibit a common variation of cleavage fracture toughness with temperature, gave birth to the notion of a "master" fracture toughness transition curve for all ferritic steels.

Recently Master Curve technology has been incorporated into ASTM and ASME codes and standards. In 1997 ASTM adopted standard E1921 that describes how to measure an index temperature for the Master Curve, T_o. T_o locates the Master Curve on the temperature axis for the steel of interest. E1921 incorporates a modern understanding of elastic-*plastic* fracture mechanics, and so permits determination of T_o using specimens as small as a precracked CVN. In 1998 ASME published Code Cases N-629 and N-631. These Code Cases permit use of a Master Curve-based index temperature ($RT_{To} \equiv T_o + 35°F$) as an alternative to RT_{NDT}. Because RT_{To} is calculated from fracture toughness data, it consistently positions bounding K_{IC} and K_{IR} curves relative to fracture toughness data for all material and irradiation conditions encountered in nuclear RPV service. Such consistency cannot be achieved via the correlative RT_{NDT} techniques used currently.

Motivation for a Improved Accuracy

Price deregulation of the electric power industry in the United States fundamentally changes the economics of continued of nuclear power plant (NPP) operation. Before deregulation NPPs, which provide primarily baseload, were paid based on capacity. Now NPPs must compete with other energy sources, so utility executives are considering new operational scenarios, some of which were unheard of as little as five years ago: extending the licensed life of the plant beyond 40 years, removal of flux reduction, up-rating of the reactor, etc. These actions all increase the rate of embrittlement, causing current licensing limits to be approached at an earlier date. Also, the lead time needed to bring replacement power sources (e.g. gas turbines, coal, or license renewal of the NPP for an additional 20 years) on-line push back by nearly a decade from EOL the date on which utilities, and consequently the NRC, must make the decisions and do the analysis that decide the future of a NPP. In combination, these factors suggest that the fate of nearly 30% of currently operating pressurized water reactors (PWRs) will be decided between 2005 and 2010. Consequently, both the industry and the NRC are now considering refinement of the procedures used to estimate of RT_{NDT} at EOL with an eye to reducing known over-conservatisms while adequately protecting the public safety. Use of the Master Curve is but one of these refinements

In addition to these economic motivations for change, regulatory motivations exist as well. The perception, based on RT_{NDT}, of a lower toughness RPV than actually exists restricts unnecessarily the permissible pressure-temperature envelope for routine heat-up and cool-down operations. This restriction increases the difficulty of performing these operations, thereby increasing the overall plant risk due to the increased probability for pump trips, and due to the increased time spent in transient (*vs.* steady state) conditions. This situation is clearly at variance with the NRC's mission of maintaining public safety. Also, the continued use of RT_{NDT} approaches, which by definition produce bounding estimates of fracture toughness, is inconsistent with the NRC's goal of moving toward a risk informed framework for rule and decision making. This framework, and the probabilistic risk assessment (PRA) methodologies that support it, require the use of best estimate values rather than bounding values whenever possible. The Master Curve provides best estimates of fracture toughness whereas RT_{NDT} technology provides bounding values, suggesting that the Master Curve fits better within a risk informed framework than does RT_{NDT}.

OBJECTIVE

In a recent NUREG, the Staff examined the technical basis for both the Master Curve itself, and for its application to the assessment of nuclear RPV integrity against fracture [Kirk 00e]. Here we focus attention on the application issues that need to be addressed to transition from the current bounding approach to toughness estimation toward a best-estimate approach that is more consistent with a risk-informed decision making process. To establish the baseline against which progress to this goal is measured, we begin by reviewing the origin of conservatisms inherent to the current RT_{NDT} / K_{IC} procedures for fracture toughness characterization.

CURRENT PROCEDURE TO ESTIMATE THE FRACTURE TOUGHNESS OF NUCLEAR RPV STEELS

Procedure Description

In all calculations to assess the integrity of a nuclear RPV against fracture, an estimate of the fracture toughness of the vessel after neutron embrittlement is needed. Practical limitations regarding the volume of material that can be irradiated as part of a surveillance program restrict both the quantity and size of the material samples used to obtain this estimate. Currently the fracture toughness of an RPV steel is estimated as follows:

1. The transition temperature of the material before irradiation ($RT_{NDT(u)}$) is determined using either ASME NB-2331 procedures [ASME NB2331], or alternative procedures intended to be conservative to NB-2331 [NRC MTEB5.2].
2. $RT_{NDT(u)}$ is shifted to account for the effects of neutron irradiation. The shift added is the difference in the CVN 30 ft-lb transition temperature (ΔT_{30}) before and after irradiation. ΔT_{30} may be either based on shift measurements (from a ASTM E185 qualified surveillance program) or on shifts calculated from chemical composition using an embrittlement trend curve [NRC RG199R2].
3. Margins are added to account for uncertainties in the state of knowledge of the material, and for uncertainties in the calculational process [NRC RG199R2, Randall 87].
4. The estimated transition temperature of the vessel after some amount of neutron irradiation (now $RT_{NDT(u)} + \Delta T_{30} +$ Margin) is used as an index temperature for the ASME K_{IC} and/or K_{IR} curves, thus establishing the lower bound above which the actual fracture toughness of the material is expected to lie.

It should be noted that nowhere in this process is the fracture toughness of the material actually measured, rather it is inferred through a series of correlations. The components of this procedure began to be established as early as 1972, and the procedure was solidified in concept as early as 1977 (NRC RG199R1). Two state-of-knowledge limitations that existed in this timeframe necessitated adoption of a correlative approach to toughness estimation:

1. <u>Linear Elastic Characterization of Fracture Behavior</u>: Between 1972 and 1977, the only mathematical description of fracture behavior sufficiently well developed for ASME codification was one premised on a linear elastic characterization of material constitutive behavior. At temperatures in fracture mode transition, large fracture toughness specimens (minimum lineal dimension of ≈2-in.) of nuclear RPV steels need to be tested to meet the validity requirements of a linear elastic fracture theory [ASTM E399]. It is not practical to use specimens of this size as part of a surveillance program.
2. <u>Need to Determine the Entire Transition Curve</u>: Calculations of the fracture integrity of a nuclear RPV require as input the complete variation of toughness with temperature through transition, not just the toughness at a fixed temperature. Between 1972 and 1977 there was no procedure available from which such a comprehensive description of transition fracture toughness behavior could be inferred based on tests of a limited number of specimens.

While approximate, the K_{IC} / RT_{NDT} procedure is believed to be, and indeed *must* be, conservative (i.e. always underestimate the measured fracture toughness of the material in question) due to the factors discussed further in the following section.

Conservatism of Procedure

Due to the LEFM Representation of Fracture Toughness

In 1972, ASME adopted the K_{IC} and K_{IR} curves to describe the variation with temperature of the static and dynamic (respectively) fracture toughness of nuclear RPV steels [WRC 175, Marston 87]. These curves were hand-drawn as lower bounds to a set of fracture toughness data valid according to the LEFM requirements of ASTM E399 [ASTM E399].

ASTM E399 places severe restrictions on the size of the plastic zone at fracture relative to the overall size of the specimen to ensure that a linear elastic description of material flow behavior is not violated in a significant way. The E399 size requirement is as follows:

$$a, b, B \geq 2.5\left[\frac{K_I}{\sigma_y}\right]^2 \qquad (1)$$

where *a* is the crack length, *b* is the length of the uncracked ligament, *B* is the specimen thickness, K_q is the stress intensity factor at fracture, and σ_y is the yield strength at the test temperature. Considering that the diameter of the plastic zone ahead of a deforming crack in a thick structure can be expressed as follows:

$$d_{plastic} = \frac{1}{3\pi}\left[\frac{K_I}{\sigma_y}\right]^2 \qquad (2)$$

one concludes that E399 requires that the smallest length scale in the specimen (*a*, *b*, or *B*) must exceed the size of the plastic zone by a factor of approximately 25 (=2.5·3·π). This restriction invariably admits only the lowest part of the population of cleavage fracture toughness values to further analysis, as illustrated in Fig. 1. Since the K_{IC} and K_{IR} curves were based exclusively on these low fracture toughness values, it is clear that the requirement for LEFM validity forces establishment of a low bounding curve.

Due to the Use of RT_{NDT} to Normalize Temperature

When using fracture toughness data to establish the bounding K_{IC} and K_{IR} curves, the fracture toughness values were not plotted *vs.* temperature, but rather *vs.* the difference between the test temperature and an index temperature called RT_{NDT} [WRC 175, Marston 78]. RT_{NDT} is determined from Charpy V-Notch (CVN) and nil-ductility temperature (NDT) data as per ASME NB-2331, as follows:

$$RT_{NDT} = MAX\{T_{NDT}, T_{35/50} - 60 \quad \text{(in °F)} \qquad (3)$$

where T_{NDT} is the nil-ductility temperature determined by testing NDT specimens as per ASTM E208, and $T_{35,50}$ is the transition temperature at which Charpy-V notch (CVN) specimens tested as per ASTM E23 exhibit at least 35 mills lateral expansion and 50 ft-lbs absorbed energy. RT_{NDT} is intended to account for the heat-to-heat differences in fracture toughness transition temperature, and thereby collapse all of the transition toughness curves for specific heats of steel onto a single curve [ASME NB2331, ASTM E208, ASTM E23]. This procedure of using RT_{NDT} to normalize temperature conservatively places the K_{IC} curve relative to measured fracture toughness data for the following reasons:

1. The NB-2331 Procedure for Determining RT_{NDT}: This procedure requires first that T_{NDT} be established, and that that three CVN tests be conducted at 60°F above T_{NDT} to demonstrate that the minimum CVN energy exceeds 50 ft-lbs, and that the minimum lateral expansion exceeds 0.035-in. NB-2331 does not require the user to either bracket the NDT temperature (i.e. achieve both break and no-break results), nor does it require determination of the temperature at which the 50 ft-lbs / 35 mil criteria is just exceeded.

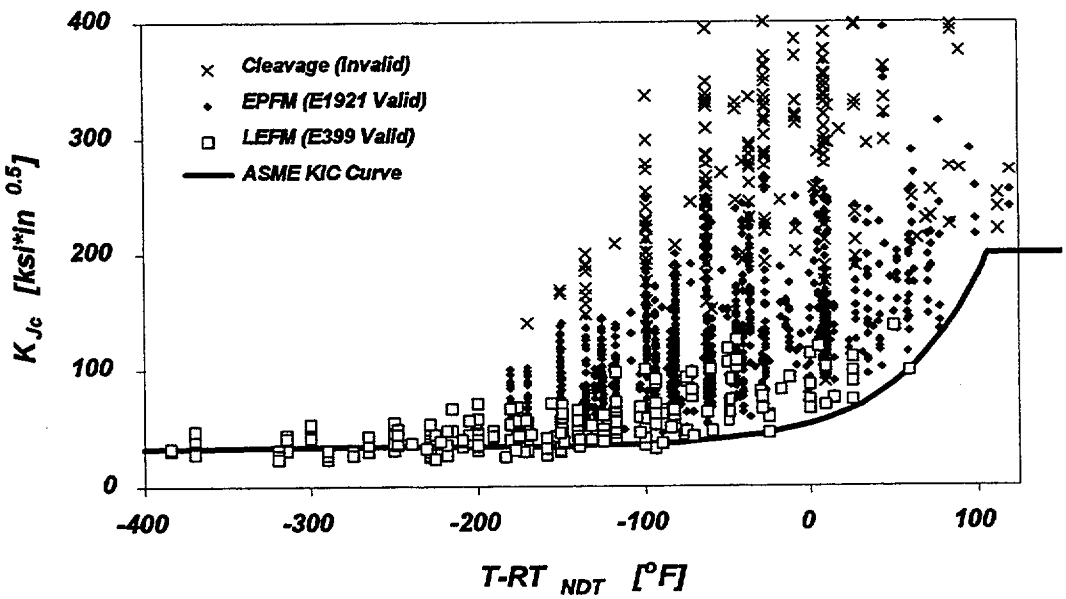

Figure 1. Placement of LEFM (ASTM E399) valid data relative to the overall population of cleavage fracture toughness data for nuclear RPV steels. All values are plotted as-measured, and are normalized relative to an ASME NB-2331 value of RT_{NDT}.

Consequently, the NB-2331 procedure forces reported values of RT_{NDT} toward the upper end of all RT_{NDT} values for a particular heat of steel.

2. The Procedure by which the Relationship Between the ASME K_{IC} curve and RT_{NDT} was Established: In the early 1970's an ASME task group established the following relationship between RT_{NDT} and the K_{IC} curve:

$$K_{IC}=33.2+2.81\cdot\exp[0.0198(T-RT_{NDT}+100)] \qquad (K \text{ in ksi}\sqrt{\text{in}},\ T \text{ in °F}) \qquad (4)$$

This equation (a hand-drawn curve at the time) was constructed in 1972 such that no existing measured K_{IC} value in transition (i.e. at T-RT_{NDT} > 100°F) fell below the K_{IC}

curve[*]. This empirical approach to developing a transition toughness curve was needed because at the time no theoretical basis existed to account for the differences in loading, loading rate, crack geometry, and specimen thickness between NDT and CVN tests and the conditions of interest in nuclear RPV service (i.e. a sharp crack in a thick structure).

The substantial collection of fracture toughness data available today (Fig. 1) testifies to the bounding characteristics achieved through the use of the ASME NB-2331 definition of RT_{NDT} along with the ASME K_{IC} curve[†]. It is important to recognize that the *combined* effects of these two factors produce a bounding curve. Neither the ASME NB 2331 definition of RT_{NDT} nor the ASME K_{IC} equation acting individually ensures bounding.

Quantification of Conservatism

Because the index temperature RT_{NDT} is determined with complete independence from the fracture toughness data it represents through its use with the ASME K_{IC} curve (eq. (4)), there is no guarantee that, for example, a K_{IC} curve positioned with respect to RT_{NDT} will always under-estimate K_{IC} data by the same amount. In fact, quite the contrary is true, as illustrated in Fig. 2. Recently, Bass et al. [Bass 00] quantified the range of possible conservatism inherent to a K_{IC} curve positioned using RT_{NDT} by the procedure illustrated in Fig. 3. Fig. 4 shows the results of this analysis, which demonstrate that a definition of the transition temperature that consistently positions a bounding curve relative to fracture toughness data can fall below RT_{NDT} by up to 200°F, illustrating the conservatism inherent to the RT_{NDT} process.

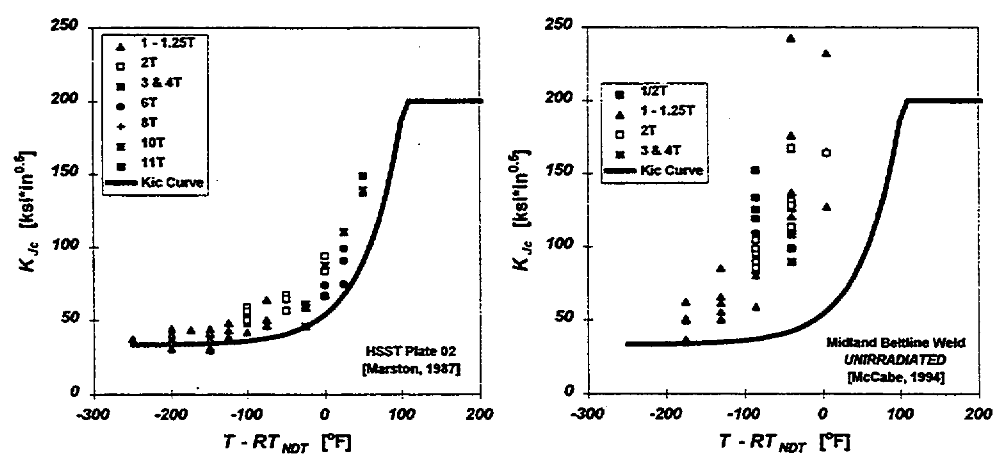

Figure 2. Illustration of the inconsistency with which RT_{NDT} positions the K_{IC} curve relative to as-measured fracture toughness data.

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[*] The ASME committee did not enforce this bounding requirement on the lower shelf, as evidenced by the considerable number of K_{IC} values that fall below the 33.2 ksi√in asymptote in Fig. 2(a).

[†] Only one K_{IC} value falls below the K_{IC} curve in transition. A K_{IC} value of 98¼ ksi√in measured using a 6T C(T) of HSST Weld 72W falls 0.9 ksi√in below the ASME K_{IC} curve at $T\text{-}RT_{NDT}$ = +59.4°F.

APPLICATION OF THE MASTER CURVE IN RPV INTEGRITY ASSESSMENT

Table 1 summarizes the codes, standards, and regulations that concern estimation of fracture toughness values used in nuclear RPV integrity calculations. The first two steps identified in Table 1 include a standard to measure toughness, and a procedure that uses this information to position a reference toughness curve on the temperature axis. ASTM E1921-97 and ASME

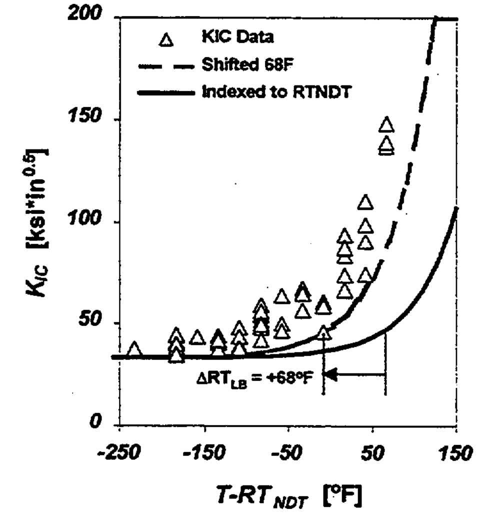

Figure 3. Procedure for defining the conservatism inherent to a K_{IC} curve located based on RT_{NDT} relative to measured K_{IC} data for the same steel [Bass 00]. The RT_{NDT}-located K_{IC} curve is translated toward the dataset until it intersects the first K_{IC} value in transition. The amount of translation defined ΔRT_{LB}.

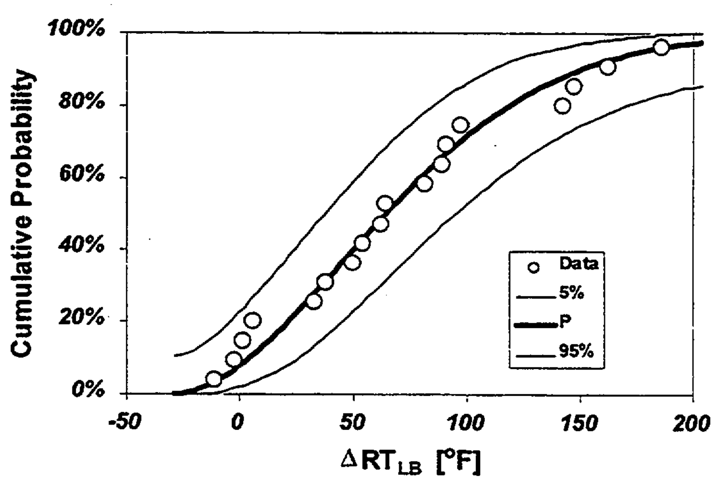

Figure 4. Conservatism inherent to a K_{IC} curve located based on RT_{NDT} quantified by applying the procedure illustrated in Fig. 3 [Bass 00] to and expanded set of LEFM valid data assembled by the Oak Ridge National Laboratory (ORNL) [Bowman 00].

Code Cases N-629 and N-631 fulfill these needs for the Master Curve. Questions raised previously by the Staff regarding the use of Master Curve technology in these codes and standards [Mayfield 97, Kirk 00a] have received considerable attention over the past few years, and are now largely resolved [Kirk 00e]. These questions, and the resolution status of each, are as follows:

1. ASTM E1921-97
   a. Is the single temperature dependence of the Master Curve appropriate for all RPV steels of interest, even after irradiation?: On-going research activities performed by both Natishan (and co-workers) [Natishan 98, Natishan 99a, Natishan 99b, Wagenhofer 00a, Wagenhofer 00b, Kirk 00b] and Odette (and co-workers) [Odette 00] provide encouraging evidence that questions regarding the theoretical limits on the universal Master Curve shape will soon be resolved. These results provide guidance on two related questions:
      i. Breadth of Applicability: Research focused on establishing the physical basis for a universal Master Curve shape reveals that the lattice structure alone controls the temperature dependence of fracture toughness. Thus, the Master Curve will model well the temperature dependence of fracture toughness for all pressure vessels steels of any product form both before and after irradiation because all of these steels have a BCC matrix phase lattice structure.
      ii. Effect of Test Temperature: T_o values determined as per E1921-97 do not show a systematic bias or trend with test temperature, nor is this expected due to the common dependence of fracture toughness on temperature for all ferritic steels. Revisions to E1921-97 propose further restriction to the range of temperatures within which one is permitted to perform toughness tests to estimate T_o. Available empirical evidence suggests that this additional restriction is not necessary.
   b. Does the ¼-power scaling rule adopted within the Master Curve reflect appropriately the effect of specimen size on fracture toughness?: Provided the material has a random distribution of cleavage initiations sites spread homogeneously throughout its volume, the Weibull model of cleavage fracture toughness in transition relies only on the existence of a state of small scale yielding to ensure its theoretical applicability. As the micro-scale inhomogeniety needed to violate the assumption of a random distribution of cleavage initiation sites is not characteristic of RPV steels, applicability of the Master Curve statistical fracture model can be assessed based on a calculation of the deformation state at fracture. Under small scale yielding conditions, fracture toughness will scale with thickness raised to the ¼-power. This result is anticipated theoretically and is well confirmed experimentally.
   c. Are T_o values determined using precracked CVN specimens equivalent to T_o values determined using larger specimens?: T_o values determined using precracked CVN specimens show a systematic bias relative to T_o values determined using physically larger samples. This bias depends on the deformation level at fracture. Information is presented herein that can be used to correct for this bias. It is important that such a correction be reviewed and balloted by ASTM committee E08 due to the interest of nuclear licensees in using precracked CVN specimens removed from surveillance to estimate T_o.

2. ASME Code Cases N-629 and N-631

a. Will K_{IC} and K_{IR} curves indexed using T_o provide an equivalent implicit margin to current approaches?: These Code Cases provide a Master Curve-based index temperature for the K_{IC} and K_{IR} curves that produce implicit margins functionally equivalent to those historically accepted for RT_{NDT}. The relationship between RT_{To} and T_o, i.e. $RT_{To} \equiv T_o + 35°F$, is defensible as it bounds a reasonable percentage of all fracture toughness data now available (97.5%) for a crack front length (2.1-in.) that exceeds the great majority of flaws found in RPV fabrication.

In contrast to this substantial progress, Steps 3 and 4 in Table 1 have received little focus to date. Nevertheless, plant-specific Master Curve submittals have moved / are moving forward. In the next section we summarize these submittals, and discusses how each submittal has addressed Steps 3 and 4 in Table 1, both of which go beyond the scope of ASTM and ASME codes and standards. This discussion is followed by a section concerning the essential characteristics a general framework to estimate the fracture toughness at EOL. Finally, we discuss recent progress, or lack thereof, toward developing the various components of such a general framework.

Plant-Specific Applications of Master Curve Technology

To date the commercial nuclear power industry has brought two submittals before the NRC that use the Master Curve to estimate the vessel fracture toughness at EOL and assess compliance with 10CFR50.61 (i.e., with the PTS Rule). These submittals concerned / concern the licenses of the Zion [Yoon 95] and Kewaunee [Lott 99, Lott 00, Server 00] NPPs‡:

- Zion: In the Zion submittal the licensee sought to use Master Curve technology and fracture toughness data on the limiting vessel material (Linde 80 weld WF-70) to establish a new un-irradiated value of RT_{NDT} [Yoon 95]. The protocols of 10CFR50.61 were then used to estimate the effects of both irradiation and uncertainties on this value, and to establish a PTS screening criteria to compare this value to. The Zion submittal did not modify 10CFR50.61 protocols to account for the use of Master Curve technology to estimate RT_{NDT}.
- Kewaunee: In a series of papers concerning the Kewaunee submittal, Lott, et al. outline several strategies to use measured values of T_o, both un-irradiated and irradiated, to estimate a RT_{NDT}-like quantity at EOL [Lott 99, Lott 00, Server 00]. In developing these estimation strategies, the authors sought to use T_o to estimate a RT_{NDT}-like quantity in a manner that parallels and satisfies the intent of current regulations (i.e. 10CFR50.61). In the Kewaunee submittal this RT_{NDT}-like quantity was compared to the current PTS screening criteria [10CFR50.61]

In summary, lacking any established alternative approach, the Zion and Kewaunee submittals both align closely with current procedures to estimate the toughness for some future irradiation condition, and to assess the adequacy of this toughness during a postulated PTS event. This approach invariably leads to assignment of burdensome margins to account for mis-fits, both

‡ Since the NRC's response to the Kewaunee submittal is still pending, a detailed discussion is not appropriate at this time. Consequently, reference is made only to information presented at ASME conferences concerning the Kewaunee submittal.

real and perceived, between Master Curve technology and the 10CFR50.61 framework. We examine the potential for moving away from this paradigm in the next section.

Table 1. Codes, Standards, and Regulations that Govern the Assessment of Fracture Toughness for Use in a PTS Analysis.

	Step	Current Technology	Master Curve Technology
1	**Measure a Material Property**	*CVN*: ASTM E23 *NDT*: ASTM E208	T_o: ASTM E1921
2	**Establish an Index Temperature and Define a Reference Toughness Curve**	RT_{NDT} ASME NB-2331	RT_{To} ASME N-629 and N-631
3	**Estimate the Toughness of Some Future Irradiation Condition (e.g., at EOL)**	Expressed in: 10CFR50.61, 10CFR50 APPG, ASME XI-G Based on: SECY 82-465, NRC MTEB5.2, NRC MEMO 82, Randall 87	Not Yet Established
4	**Establish a Screening Criteria for PTS**	Expressed in: 10CFR50.61 Based on: SECY 82-465	Not Yet Established

Progress Toward a Generic Master Curve Methodology

The information presented in Table 1 points out that factors exist beyond those considered thus far by ASTM and ASME that need to be addressed to bring Master Curve technology to the point that it can be applied routinely to assess nuclear RPV integrity:

1. Procedures to estimate the toughness at EOL: These procedures would predict T_o and/or RT_{To} for future irradiation conditions from available information (i.e. mechanical properties, chemical properties, fluence), and adjust these estimates to account for various uncertainties. Toughness is determined through the association of these index temperatures with fracture toughness transition curves. Reg. Guide 1.99 Rev. 2 describes the procedures used currently to this end [NRC RG199R2]§. No parallel rule or guidance exists currently for Master Curve-based methodologies.

§ These procedures find their origins in the work that led up to and provided the technical basis for the current PTS screening criteria [NRC MTEB5.2, NRC MEMO 82, Randall 87, SECY 82-456]. Nevertheless, the procedures are *applied* to estimate toughness not only for use in a PTS assessment (where 10CFR50.61 adopts Reg. Guide 1.99 Rev. 2 procedures and applies them at EOL fluence), but also as part of the calculations that establish heat-up and cool-down limits for routine operation [10CFR50 APPG, ASME XI-G].

2. <u>A PTS screening criteria</u>: This would be a value / values to which a Master Curve-based estimate of T_o and/or RT_{To} at EOL would be compared to assess the suitability of the reactor for operation through EOL. SECY-82-465 establishes the technical basis for the current criteria (300°F for circumferential welds, 270°F for longitudinal welds, plates, and forgings) of 10CFR50.61 [SECY 82465, 10CFR5061]. No parallel rule or guidance exists for the Master Curve.

In this section we examine the current RT_{NDT}-based procedure to estimate the fracture toughness at EOL and discuss its role in establishing the current PTS screening criteria. This discussion provides a perspective on the obstacles that plant specific Master Curve applications have encountered in attempts to parallel current procedures. In the following sections we turn attention toward the future research and development achievements needed to eliminate these obstacles.

The model used to estimate toughness in the PFM calculations that established the current PTS screening criteria is as follows [SECY 82465]:

$$RT_{NDT(f)} = RT_{NDT(u)} + \Re \cdot \Delta RT_{NDT(f)} \tag{5}$$

where

- $RT_{NDT(f)}$ is the estimated RT_{NDT} of the vessel material after irradiation to the fluence f. Toughness is determined from $RT_{NDT(f)}$ through its use as an index temperature for the K_{IC} and K_{IR} curves
- $RT_{NDT(u)}$ can represent ***either*** of the following values:
  - A value of RT_{NDT} in the unirradiated condition based on testing a specific vessel material in accordance with ASME NB-2331, ***or***, if such measurements are unavailable,
    - <u>For Welds</u>: A generic mean value determined from a data set relevant to the material class of interest. Currently accepted generic mean values include -56°F for welds made with Linde 0091, 1092, 0124, and ARCOS B-5 welding fluxes, and -5°F for welds made with Linde 80 flux.
    - <u>For Plates</u>: If only CVN data are available, as is sometimes the case for plate materials, MTEB-5.2 provides procedures to estimate RT_{NDT} values that are intended to be conservative to (i.e. higher than) RT_{NDT} values determined using ASME NB-2331 [NRC MTEB52].
- $\Delta RT_{NDT(f)}$ is the mean value of the irradiation induced transition temperature shift, and is calculated as follows:

$$\Delta RT_{NDT(f)} = (CF) f^{(0.28 - 0.10 \log f)} \tag{6}$$

$\Delta RT_{NDT(f)}$ can represent ***either*** of the following values:

- It is the mean value of the of this shift for the material samples tested as part of the credible surveillance program, ***or***, if the surveillance data is not deemed to be credible,
- It is the mean value of this shift for a material having the composition (Cu and Ni) corresponding to the heat average for the entire heat of material in question.

In the former case, when credible surveillance data is used to establish ΔRT_{PTS}, the value $\mathfrak{R}$ adjusts ΔRT_{PTS} to account for differences between the chemical composition of the surveillance material and the heat average chemical composition. $\mathfrak{R}$ represents the "ratio procedure" as described in 10CFR50.61(2)(ii)(B). $\mathfrak{R}$ is defined as the chemistry factor (*CF*) for the best estimate composition of the heat divided by the chemistry factor for the specific composition of the surveillance weld. Tables in 10CFR50.61 define chemistry factors based on material product form, Cu, and Ni.

Natishan and co-workers have recently developed a diagrammatic representation of eq. (5), Fig. 5, which illustrates how the value of an input parameter (e.g. Cu, Ni, ϕt, CVN, NDT, etc.) "flows" through eq. (5) to produce an estimate of the value of RT_{NDT} after irradiation to EOL fluence [Li 00]. Thus, in addition to its use in determining the PTS screening criteria, eq. (5) also establishes the variability in estimates of $RT_{NDT(f)}$ that are compared to this screening criteria. This amount of variability, often called a "Margin," is traditionally added to the estimate of $RT_{NDT(f)}$ as follows [NRC RG199R2]:

$$RT_{NDT(f)} = RT_{NDT(u)} + \mathfrak{R} \cdot \Delta RT_{NDT(f)} + M \tag{7}$$

$$M = 2\sqrt{\sigma_I^2 + \sigma_\Delta^2} \tag{8}$$

where

- σ_I is the standard deviation in the value of $RT_{NDT(u)}$. It can represent ***either*** of the following values:
  - σ_I is *"determined from the precision of the test method"* if $RT_{NDT(u)}$ is established either (a) by testing the specific vessel material in accordance with ASME NB-2331, or (b) by MTEB-5.2 procedures. While not explicitly stated in 10CFR50.61, a value of σ_I = 0°F is used in this situation.
  - If a measured value of $RT_{NDT(u)}$ is not available, σ_I is the standard deviation of the data set used to establish the generic mean value of $RT_{NDT(u)}$. The most common value of in this situation is 17°F [NRC MEMO 82]. This value applies to welds made with Linde 0091, 1092, 0124, ARCOS B-5, and Linde 80 welding fluxes. Other values, like 26.9°F for B&W plate materials have also been established and are recorded in RVID.

  In both cases the sum $\{RT_{NDT(u)} + 2\sigma_I\}$ represents a bounding value of RT_{NDT} before irradiation. When RT_{NDT} is determined according to ASME NB-2331 or MTEB-5.2, these protocols produce a bounding estimate, so σ_I can be zero. However, when a mean value of RT_{NDT} is used then $2\sigma_I$ =34°F needs to be added to produce a bounding estimate.
- σ_Δ is the standard deviation in the value of ΔRT_{PTS}. It can represent ***either*** of the following values:
  - If credible surveillance data **is not** available, the σ_Δ values are 28°F for welds and 17°F for plates
  - If credible surveillance data **is** available, the σ_Δ values are 14°F for welds and 8.5°F for plates.

These observations illustrate that the main difficulty faced by plant specific Master Curve applications has been the lack of an accepted framework by which to estimate the irradiated fracture toughness of the vessel from T_o data (i.e. a version of eq. (5) for T_o), and the fact that this framework was never used to establish a PTS screening criteria for T_o. Consequently, there is currently no T_o-based PTS screening criteria, and there is no T_o-based margin term (i.e. an eq. (8) for T_o) based on uncertainty in the input variables. Beyond these general difficulties, the Zion and Kewaunee submittals have encountered certain specific concerns in their attempts to parallel eqs. (7) and (8), as follows:

1. Zion: If an un-irradiated T_o is used and shifted using the Reg. Guide 1.99 Rev. 2 fluence function, concerns have arisen regarding the appropriateness of applying a CVN-based shift to fracture toughness data.
2. Kewaunee: With the current methodology, the toughness after irradiation can only be estimated from the sum of an un-irradiated reference temperature and an irradiation-induced shift in the reference temperature. Direct measurement of the irradiated transition temperature was not considered when the calculations that support the current PTS rule were adopted. Consequently, this approach currently lacks an established basis to account for differences between the composition of the surveillance samples and the composition of the material in the vessel. The existing Ratio procedure operates on the irradiation-induced shift in the transition temperature, not on its absolute value, making the proper application of this procedure to an irradiated transition temperature unclear.

Ultimately there is the nagging concern that forcing Master Curve-technology into the current, non-Master Curve, framework may produce systemic "lack of fit" uncertainties, thereby resulting in the need for higher margins. The only way to alleviate this concern is to establish a Master Curve framework to estimate toughness at EOL, and use this framework as part of the PFM calculations to establish a PTS screening criteria applicable specifically to Master Curve-based estimates of fracture toughness. Work on the development of such a framework for the Master Curve has only recently begun [Natishan 00]. In the following sections we review recent progress in the developing some of the components of such a framework, including:

1. Generic values of T_o for use when plant specific data is unavailable
2. Irradiation damage effects on T_o (Irradiation trend curves)
3. Treatment of the newly recognized linkage between fracture toughness and crack front length.
4. Treatment of the loading rate effect on fracture toughness to establish the position of the crack arrest curves used in establishing the PTS screening criteria.

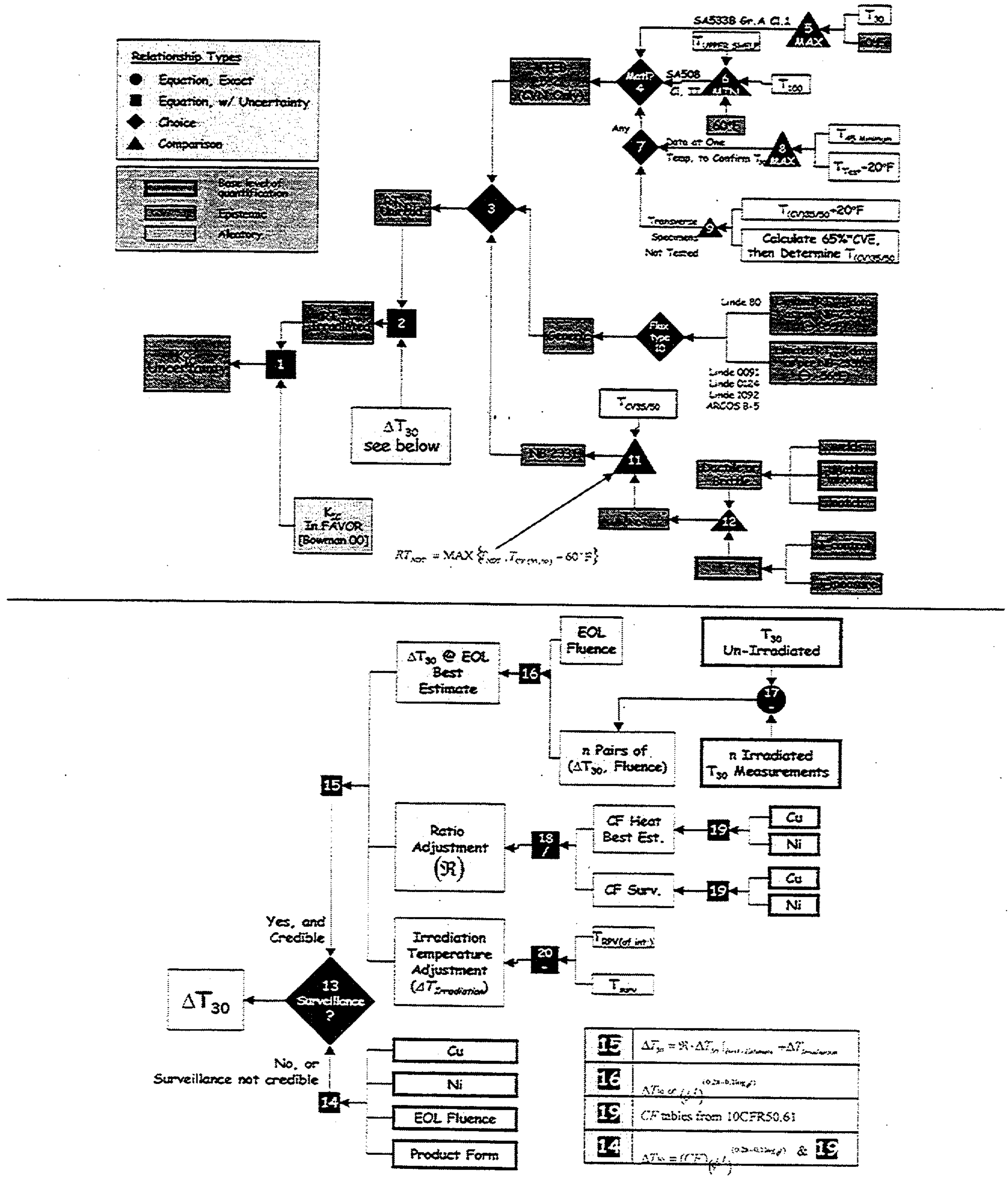

Figure 5. Root cause diagram illustrating the methodology used currently to estimate the fracture toughness of a reactor pressure vessel steel after irradiation [Li 00].

Generic Values of T_o

Current RT_{NDT}-based procedures provide generic values of un-irradiated RT_{NDT} for use when material specific information is not available. Similar generic values of RT_{To} will most likely be needed as part of a Master Curve methodology that is usable by all plants. Here we use a large collection of fracture toughness values [Rosinski 99] to establish candidate generic RT_{To} values by the following procedure:

1. The database is queried to identify all fracture toughness data available for a particular class of RPV materials. Here we consider classes defined by flux type (for welds) and by ASTM material specification (for plates and forgings).
2. The fracture toughness values are normalized to a 2.1-in. thickness using the following weakest-link relationship included in ASTM E1921-97:

$$K_{Jc(2.1T)} = K_{\min} + \left(K_{Jc(measured)} - K_{\min}\right)\left(\frac{B}{2.1}\right)^{1/4} \tag{9}$$

   A "size" of 2.1-in. is selected to maintain consistency with the average size associated with the original K_{IC} database used to establish the relationship between K_{IC} data and RT_{NDT} for the current ASME K_{IC} curve [Marston 87].
3. These size-normalized fracture toughness values are plotted *vs.* test temperature. A K_{IC} curve, i.e.

$$K_{IC} = 33.2 + 2.81 \cdot \exp\left[0.0198 \cdot \left(T - RT_{To(generic)} + 100\right)\right], \quad (K \text{ in ksi}\sqrt{\text{in}},\ T \text{ in °F}) \tag{10}$$

is then plotted, and the value of $RT_{To(generic)}$ is adjusted position the curve so that it bounds 97.5% of the fracture toughness values in fracture mode transition. While in principal any tolerance bound can be selected, we selected a 97.5% value to maintain consistency with how a RT_{To} positioned K_{IC} curve bounds the original K_{IC} data set [Wallin 97].

Fig. 6 illustrates this procedure for A533B Cl. 1 plate and for Linde 80 welds, while Table 2 summarizes $RT_{To(generic)}$ values for the different RPV material classes. This procedure to establish generic values of RT_{To} incorporates the material uncertainty within the class into the value of $RT_{To(generic)}$ by basing the position of the 97.5% tolerance bound curve on fracture toughness data for a number of different heats from the same material class. Consequently, if these values of $RT_{To(generic)}$ are used in a plant assessment, a non-zero uncertainty term (equivaluent to σ_I in the current methodology) should ***not*** be used.

Estimate of Irradiation Damage Effects on T_o

As expressed by eq. (5), the current technique for estimating the transition temperature after irradiation is to add an irradiation shift to an un-irradiated transition temperature value. The shift in the CVN transition temperature at 30 ft-lbs is currently calculated from fluence and composition using the following formula [NRC RG199R2]:

$$\Delta T_{30} = (CF) f^{(0.28 - 0.1 \log f)} \tag{11}$$

Table 2. Generic RT_{To} values for different classes of nuclear RPV materials

Material Class	$RT_{To(Generic)}$ [°F]	Total Number of K_{Jc} Values	Number of K_{Jc} Values not Bounded	% Bounded
A508 Cl. 2	-14	38	0	100.0%
A508 Cl. 3	-42	606	15	97.5%
A302B	14	58	1	98.3%
A302B Mod.	-39	26	0	100.0%
A533B Cl. 1	18	1481	36	97.6%
Linde 0091	2	71	1	98.6%
Linde 0124	-25	178	4	97.8%
Linde 1092	-151	148	3	98.0%
Linde 80	-34	213	5	97.7%

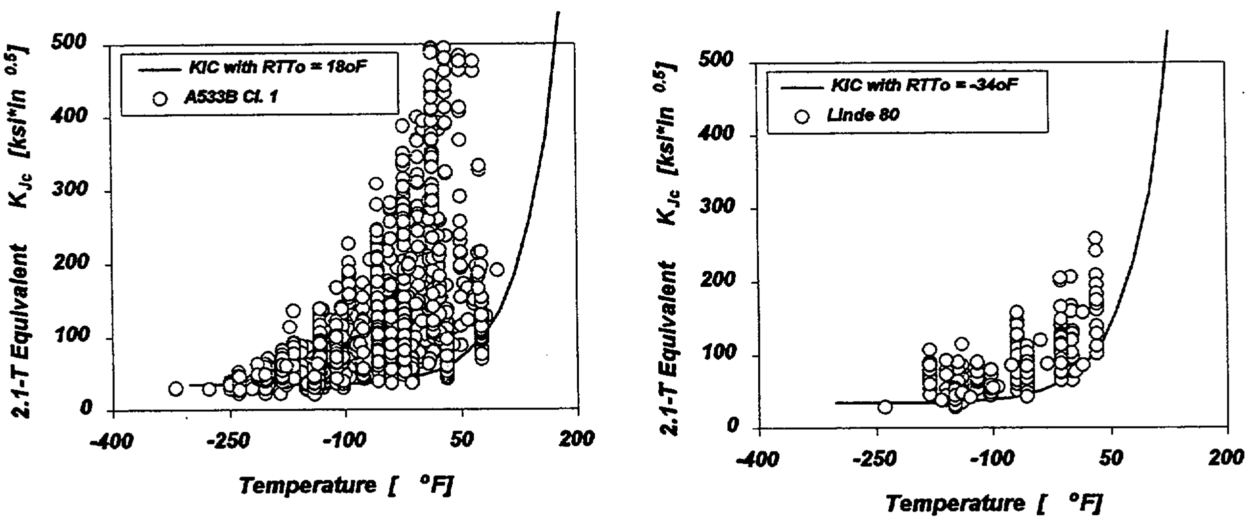

Figure 6. Use of fracture toughness data for A533B Cl. 1 (left) and for Linde 80 (right) to establish generic values of RT_{To}.

Here the *CF* (chemistry factor) expresses the aggregate effect of Cu, Ni, and product form on irradiation sensitivity. Reg. Guide 1.99 Rev. 2 includes tables of *CF* values for a range of compositions. The form of the fluence function in eq. (11), i.e. $f^{(0.28-0.1\log f)}$, was established by curve-fitting a database of 177 ΔT_{30} values [Randall 87]. In Master Curve-based applications, a question arises regarding the appropriate form of the shift equation for T_o. Since the irradiation shifts in both Charpy and fracture toughness transitions are largely controlled by increases of material flow strength produced by irradiation, it seems reasonable that the fluence function for shifts of Charpy transition temperature might model shifts in the fracture toughness transition temperature (i.e. T_o) as well. Sokolov and Nanstad compared irradiation shifts of both CVN energy and fracture toughness transition [Sokolov 96]. This comparison (see Fig. 7) showed a 1:1 correlation for welds (42 data points). Conversely, examination of 47 plate materials shows that irradiation shifts the fracture toughness transition temperature 16% more than it does the CVN transition temperature. In both cases the relationship between the two transition temperatures was linear. More recently, Onizawa and Suzuki presented results demonstrating a nearly 1:1 correlation (ΔT_o = 1.03$\cdot\Delta T_{30}$) for 4 extensively characterized plates [Onizawa 00]. Also Kirk et al. compared available data on T_o shifts produced by irradiation to the functional form of eq. (11) (see Fig. 8). Figures 7 and 8 both suggest that the Reg. Guide 1.99 (Rev. 2) fluence function provides a reasonable description of the shift in T_o produced by irradiation.

These results are encouraging. However, the high cost of irradiated material testing will likely preclude development of a sufficiently well populated database of T_o shift values to either directly develop a T_o-based irradiation trend curve, or even to test empirically the appropriateness of eq. (11) for the conditions of interest. Consequently, resolution of this issue could rest with establishing a sound basis for why T_o and CVN shifts should be the same, or at least related. Existence of such a rationale, which is not currently being investigated, would pave the way for establishing the appropriate functional form for T_o shifts based on extensive databases of CVN shifts values that are now available [Eason 98].

<u>The Effect of Crack Front Length on T_o</u>

The Master Curve incorporates the following relationship between fracture toughness and the length of the crack front based on a weakest-link model of cleavage fracture under small scale yielding conditions:

$$K_{Jc(Size1)} = K_{min} + \left(K_{Jc(Size2)} - K_{min}\right)\left(\frac{B_{Size2}}{B_{Size1}}\right)^{1/4} \tag{12}$$

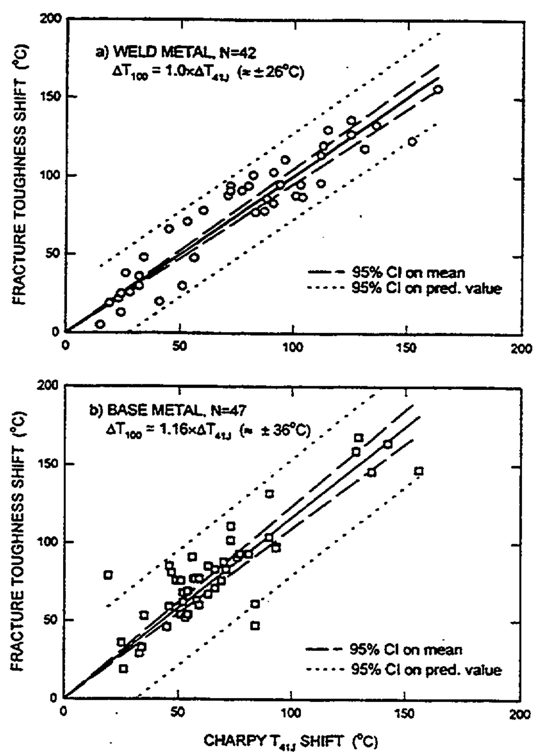

Figure 7. Comparison of irradiation induced CVN and T_o shifts for nuclear RPV welds and base materials [Sokolov 96].

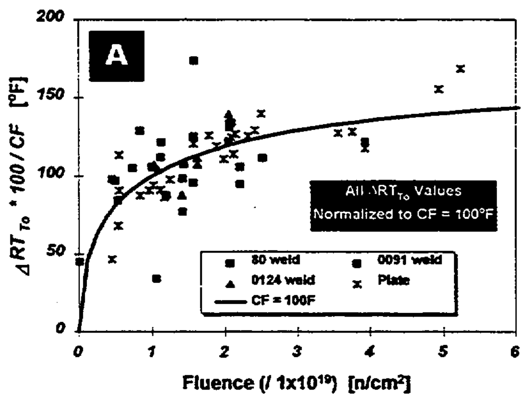

Figure 8. Variation of ΔRT_{To} with fluence, and comparison with Reg. Guide 1.99 Rev. 2 fluence function developed to describe trends in Charpy V-Notch data [Kirk 99].

Here, K_{min} = 20 MPa√m and represents the value of applied K_I below which cleavage fracture is not possible. The subscripts "Size1" and "Size2" refer to toughness (K) or thickness (B) values for two different specimen thicknesses. Eq. 12 applies to straight-fronted cracks in a state of small scale yielding. It predicts a decline in fracture toughness with increasing crack front length, a prediction in accord with considerable experimental evidence for fracture test specimens [Kirk 98a, Rathbun 00]. Eq. (12) represents a significant departure from current ASME code practice that treats toughness and crack front length as independent variables.

The practice of positioning a bounding curve relative to fracture toughness data addresses the effect of crack front length on toughness. This practice implicitly links to the bounding curve the crack front length(s) characteristic of the fracture toughness data used to establish its position. Thus, both RT_{NDT} and RT_{To} indexed K_{IC} curves have the same implied crack front length because the original K_{IC} data set [Marston 87] provided the basis for positioning both curves. However, once a fracture assessment methodology used the Master Curve directly rather than just using T_o to position a bounding curve, explicit procedures to determine the effect of crack front length on fracture toughness will be needed. Since vessels contain either embedded elliptical flaws or semi-elliptical surface breaking flaws, this methodology will need to treat crack front length effects, and address their interaction with loss of constraint effects, for non-straight fronted cracks. Fig. 9(a) compares cleavage fracture toughness data for semi-elliptical surface cracks in A515 steel with a Master Curve for this material [Joyce 97b, Porr 95]. This comparison illustrates that the relationship between crack front length and fracture toughness that works so well for straight fronted cracks in fracture toughness specimens, eq. (12), places data for part-through surface cracks too high relative to the standard Master Curve. In Fig. 9(b) these data are brought into agreement with the Master Curve by using only 20% of the total crack front length of the past-through surface cracks when calculating their equivalent 1T fracture toughness.

The analysis presented in Fig. 9 is a very rudimentary. It fails to discriminate between the variable-K_I field around the crack front or loss of full constraint where the crack front intersects the free surface as the cause of this change in the scaling relationship. Nevertheless, the analysis does suggest that, whatever the cause, only a small fraction of the crack front length in a non-straight fronted crack contributes significantly to the probability of cleavage fracture. To enable application of the Master Curve to structures, a form of eq. (12) that addresses non-straight fronted fatigue cracks, and can treat both statistical size effects and constraint effects, is needed. Several on-going research programs address this goal using Weibull models coupled with 3D elastic-plastic finite element analysis to predict fracture the conditions for crack initiation from semi-elliptical surface cracks [Gao 99, Bass 00b]. Ultimately, these efforts will need to both provide a predictive model and assess the breadth of material / irradiation / loading conditions to which the model applies.

Non-straight fronted cracks are considered in reactor pressure vessel integrity analysis in the following three areas:

1. Flaw specific-assessments performed according to ASME Section XI (IWB-3500, IWB-3600),
2. PTS analysis as described in 10CFR50.61 and performed in accordance with Regulatory Guide 1.154, and
3. Calculation of permissible limits on heat-up and cool-down performed in accordance with ASME Section XI Appendix G.

In the first two cases, the flaws used in the calculations represent flaws that exist, or could exist in an operating RPV. Thus, a technical resolution of the effect of crack front length on fracture toughness should provide an appropriate analysis methodology for these calculations. Conversely, heat-up and cool-down curves are calculated for a postulated flaw that penetrates one-quarter of the way through the reactor pressure vessel wall and has a 6:1 ratio of surface breaking length to depth. This size of this flaw exceeds considerably that observed in any operating RPV (an 8-in. thick vessel this flaw would have a crack front length of 14-in.), making the flaw size a conservatism implicit to this analysis methodology. Thus, before the Master Curve can be used for Appendix G analyses, a reconciliation of the ¼-T flaw methodology and the Master Curve approach is needed.

<u>The Effect of Loading Rate on T_o</u>

Because the postulated failure of a RPV would likely involve a rapidly propagating crack, the fracture integrity assessment methodology needs to account for the effect of loading rate on fracture toughness. Rate effects enter the methodology via the separation between the static and dynamic fracture toughness curves. This separation is currently fixed irrespective of either the loading rate differential between the two curves, or the strength level / degree of irradiation of the material in question [Yoon 99]. Nevertheless, empirical evidence abounds that both loading rate and material strength influence the fracture toughness transition temperature [Barsom 87].

Currently the ASME K_{IR} curve represents the lower-bound toughness for both crack initiation at an elevated loading rate, and for crack arrest. In a PTS calculation, a vessel is not considered to have "failed" unless an initiated crack cannot be arrested [Dickson 95]. Absent a change in this definition of vessel failure, treatment of crack arrest will be part of any comprehensive RPV integrity assessment strategy. While crack initiation at elevated loading rates fits well within the Master Curve framework, the same weakest link model used to characterize crack initiation

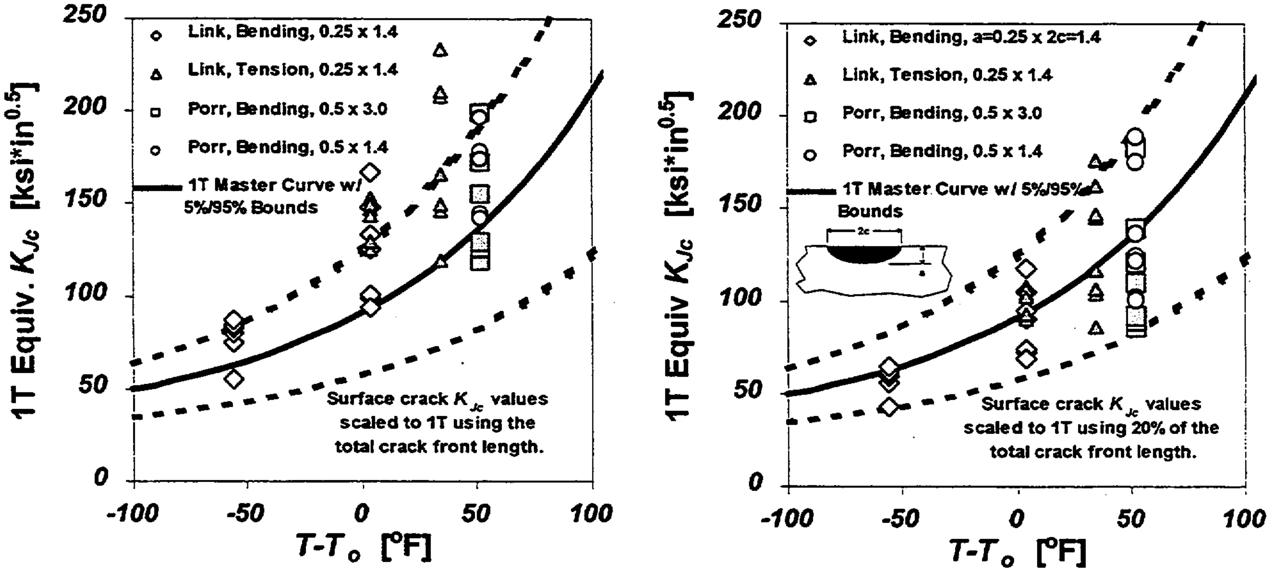

Figure 9. Comparison of data for part through surface cracks to a 1T equivalent Master Curve [Joyce 97b, Porr 95]

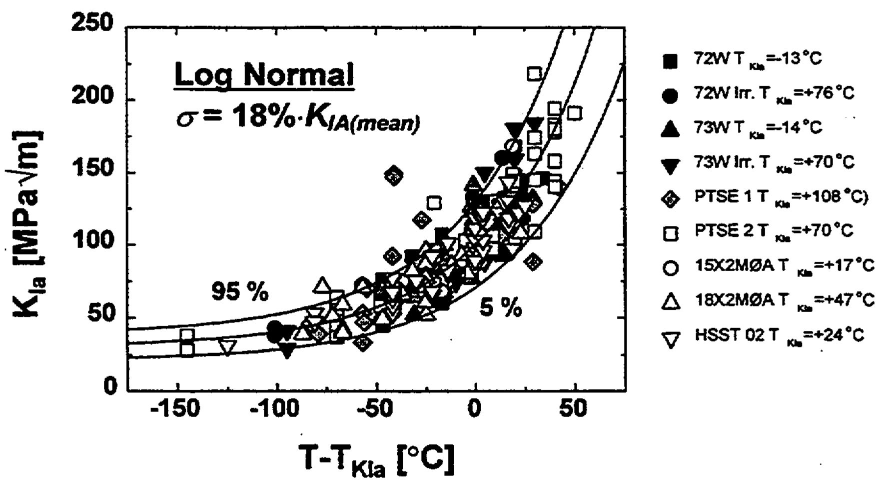

Figure 10. Crack arrest Master Curve proposed by Wallin [Wallin 98b].

clearly cannot describe crack arrest. Crack arrest will not occur until the local driving force for continued crack propagation falls below the local material arrest toughness over a significant portion of the propagating crack front [Wallin 98b]. The requirements for crack arrest are therefore controlled by a distributed process on the micro-scale, in contrast to crack initiation, which is controlled by local properties. This simple model suggests that the scatter in crack arrest toughness values should be less than for crack initiation toughness values, and that crack arrest toughness should not exhibit a statistical size effect. A recent analysis by Wallin bears out these expectations. In an examination of nine different sets of crack arrest data (seven drawn from HSST/HSSI program records) Wallin demonstrated that crack arrest data are distributed log-normally about a mean curve that has the same temperature dependence as the Master Curve (see Fig. 10).

This similarity between the temperature dependence of initiation and arrest toughness suggests the possibility of describing the position of the arrest toughness curve in terms of a shift from the position of static initiation toughness curve, e.g. as a shift relative to T_o. Wallin examined this possibility using 55 sets of data for ferritic steels that included a variety of product forms, strength grades, and irradiation conditions [Wallin 98b]. Based on a statistical analysis of these data Wallin developed the following shift equation:

$$T_{o(arrest)} - T_{o(static)} = \Delta T_{o(arrest)} = \exp\left\{4.98 - \left(\frac{T_0 + 273}{119.2}\right)^{0.915} + \left(\frac{\sigma_y}{572.4}\right)^{0.868}\right\} \qquad (13)$$

where T_o is in °C and σ_y is the static room temperature yield strength in MPa**.

ASME Code Cases N-629 and N-631 propose using RT_{To} as an index temperature for both the K_{IC} and K_{IR} curves, thereby maintaining the traditional fixed separation between these curves. Fig. 11 demonstrates that this procedure will produce a bounding estimate of crack arrest toughness provided the separation between the median curves for static initiation and crack arrest toughness falls below 95°F. In Fig. 12 we use eq. (13) determine the conditions for which separations of less than 95°F occur. This comparison is made over the range of T_o values observed for irradiated and un-irradiated RPV steels using mean yield strength values for these conditions (un-irradiated = 69 ksi, irradiated = 90 ksi) taken from the database (Appendix A). While only cursory in nature, this analysis suggests that the Code Case N-629 proposal provides a bounding curve for plants approaching their end of license (i.e. T_o > 140°F). Thus, the Code Case proposal appears to provide an adequate approach for assessment of EOL conditions (and thereby PTS).

While the correlations presented in this section provide a useful summary of the trends exhibited by available data, they cannot replace a more fundamental, physically based, understanding of why such trends should occur. The absence of such an understanding raises questions regarding the limits of applicability of these relationships, thereby impeding progress in the application of Master Curve concepts in nuclear RPV integrity assessment.

** Wallin has also published a correlation, based on analysis of 59 data sets, that permits estimation of the temperature shift between a static and dynamic crack *initiation* toughness curves [Wallin 97b]:

$$\Delta T_{o(dynamic)} = \frac{T_{o(static)} \cdot \ln(\dot{K}_I)}{\Gamma - \ln(\dot{K}_I)}, \quad \Gamma = 9.9 \cdot \exp\left\{\left(\frac{T_{o(static)} + 273}{190}\right)^{1.66} + \left(\frac{\sigma_y}{722}\right)^{1.09}\right\}$$

Differences in strain rate between crack initiation and crack arrest suggest that $\Delta T_{o(arrest)}$ will always exceed $\Delta T_{o(dynamic)}$. Furthermore, PTS events do not usually produce rapid mechanical loading rates. Consequently, we focus exclusively on crack arrest in this discussion.

SUMMARY AND CONCLUSIONS

The information provided in this paper demonstrates that substantial progress has been made recently concerning the adoption of a Master Curve testing standard, and the use of the T_o-index temperature measured by this standard to position bounding fracture toughness curves for use in vessel integrity calculations. Questions raised previously by the Staff regarding the use of Master Curve technology in these codes and standards are now largely resolved. The main difficulty faced when using the Master Curve to assess RPV integrity is now the lack of an accepted framework by which to estimate the irradiated fracture toughness of the vessel from T_o data, and the fact that this framework was never used to determine a PTS screening criteria for T_o. Consequently, there is currently no T_o-based PTS screening criteria, and there is no T_o-based margin term to account for uncertainty in the input variables. Ultimately these deficiencies fuel a concern that forcing Master Curve-technology into the current, non-Master Curve, framework may produce systemic "lack of fit" uncertainties, thereby resulting in the need for higher margins. The only way to alleviate this concern is to establish a Master Curve framework to estimate toughness at EOL, and use this framework as part of PFM calculations to establish a PTS screening criteria applicable specifically to Master Curve-based estimates of fracture toughness. Work on the development of such a framework for the Master Curve has only recently begun. In this paper we reviewed recent progress in the developing some of the components of such a framework, including the following:

1. Generic values of RT_{To} are provided for use when plant- or material-specific values of RT_{To} are not available.
2. Data is provided that demonstrates a 1:1 correlation between the irradiation shift of the Charpy-V and T_o transition temperatures. This information suggests the possibility of applying embrittlement trend curves developed from CVN data to estimate the effect of irradiation on T_o.
3. Available data suggests that weakest link scaling models developed for straight fronted cracks in test specimens systematically under-predict the fracture resistance of the semi-elliptical and buried cracks found in reactor pressure vessel service.
4. ASME Code Case N-629 uses RT_{To} to position both the K_{IC} and K_{IR} curves with a fixed temperature separation between them. Information presented in this paper demonstrates that this fixed separation under-estimates the crack arrest toughness of RPV steels in some circumstances, and over estimates it in others. This finding suggests that a revision of the Code Case is needed to ensure that the K_{IR} curve provides an appropriate degree of bounding to crack arrest data for all material conditions of interest.

These findings provide cause for optimism that the issues surrounding application of Master Curve-based methodologies to the assessment of nuclear reactor vessel safety can be favorably resolved providing focused efforts continue in a number of key areas.

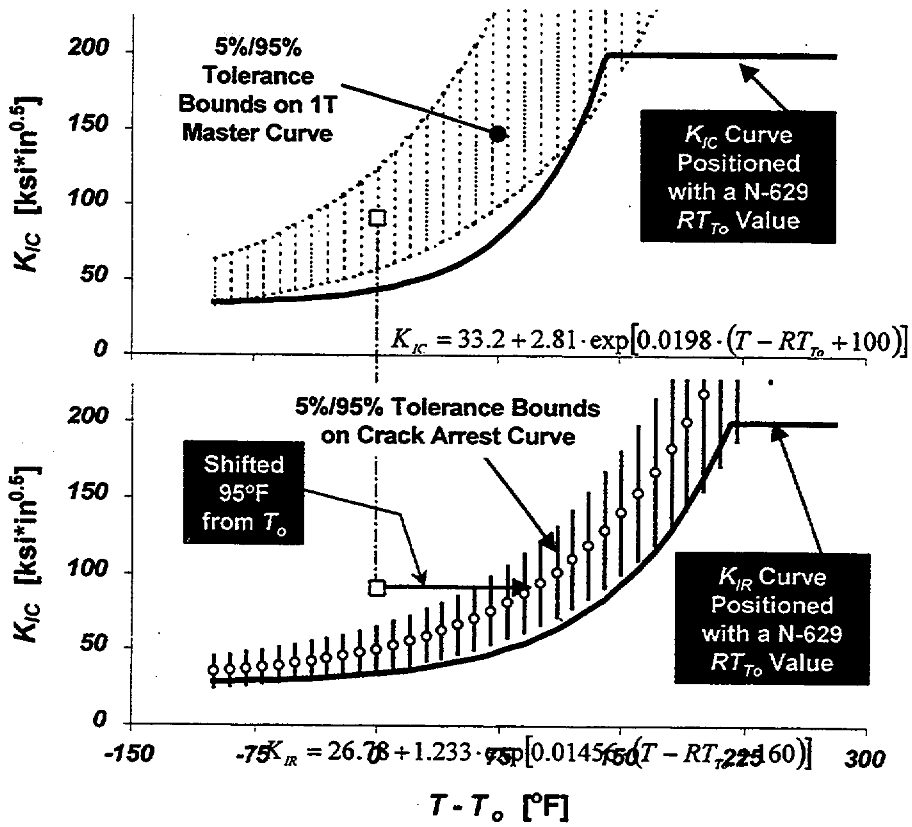

Figure 11. Illustration of the largest shift between a static initiation toughness curve (Master Curve) and a crack arrest toughness curve that will be bounded by a K_{IR} curve located using RT_{To}.

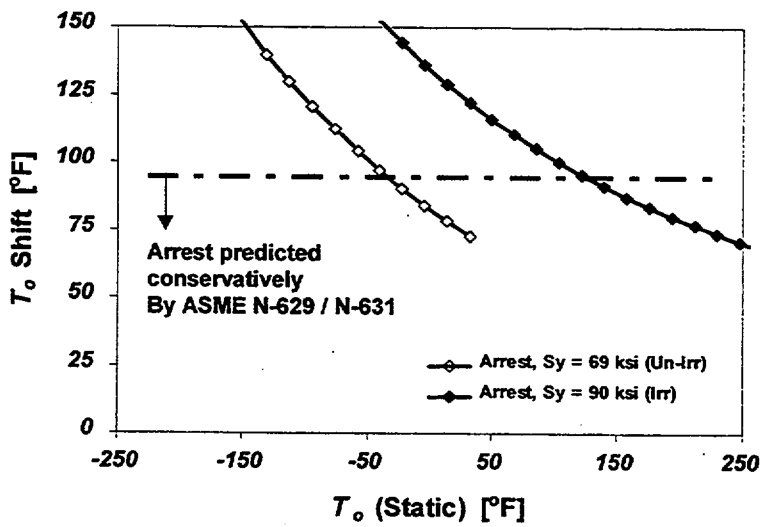

Figure 12. The shift in transition temperature between a static initiation toughness curve (Master Curve) and a crack arrest toughness curve [Wallin 98b].

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NEEDED RESEARCH TO SUPPORT DECOMMISSIONING – AN INDUSTRY PERSPECTIVE

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Abstract: In these times of reduced budgets, both Federal agencies and the private sector must optimize available resources. Each research dollar must be targeted to those proposals with the greatest potential to improve the safety and/or efficiency of nuclear technology companies.

In three important ways, the discipline imposed by limited resources, while sometimes painful, can be healthy and often yields high quality results. First, the approved research project will be focussed on solving a real problem of high priority to the organization. Second, the urge to re-invent the wheel will be controlled and research projects will be tailored to adapt and build on the huge body of excellent research available globally. Third, limited resources will encourage innovation and collaboration between government agencies and private sector resources.

In looking to the future, industry has identified decommissioning issues, pertinent to decommissioning which are explored in the subject paper.

Industry appreciates the opportunity to provide input to NRC research initiatives in conjunction with the 28th Annual Water Reactor Safety Meeting. In these times of reduced budgets, both Federal agencies and the private sector must optimize available resources. Each research dollar must be targeted to those proposals with the greatest potential to improve the safety and/or efficiency of nuclear technology companies.

In three important ways, the discipline imposed by limited resources, while sometimes painful, can be healthy and often yields high quality results. First, the approved research project will be focussed on solving a real problem of high priority to the organization. Second, the urge to re-invent the wheel will be controlled and research projects will be tailored to adapt and build on the huge body of excellent research available globally. Third, limited resources will encourage innovation and collaboration between government agencies and private sector resources.

NRC's Office of Regulatory Research has in fact accomplished just this outcome in the RESRAD revision project. Both industry and the agency recognized an important need for a dose assessment code for license termination of nuclear facilities with complex sites. The need was urgent as many facilities were in the process of developing decommissioning/license termination plans. Rather than start from scratch, NRC-RES decided to modify an existing code, RESRAD, to meet their specific needs. Finally, NRC-RES agreed to collaborate with EPRI to review the draft code. This type of approach has a greater probability of yielding a quality product within a realistic schedule and at a reasonable cost.

Looking to the future, the nuclear energy industry identified decommissioning issues, pertinent to Session 3A of this meeting, in the following areas that could benefit from additional research:

- In support of dose assessment
- In support of the GEIS supplement
- In support of material clearance rulemaking
- In support of transportation

In discussing opportunities to further support dose assessment, we are talking broadly about modeling the site, selecting dose assessment code(s), selecting and justifying the input parameters, and verifying the dose assessment output in the final status survey. The existing tools available to model reactor sites and assess the dose that could result from residual radioactivity post remediation have limitations that pose significant challenges to the industry.

They do not provide the capability to directly address sub-surface contaminated soil, current or future contaminated ground water, or contaminated sediments. Within a building, they do not support the assessment of subsurface foundations, sumps, and embedded pipe. We also need the ability to handle volumetric contamination and finite areas of inaccessible contamination. Specific guidance should be developed to address these issues with a consistent, risk informed approach.

When implementing the code, the industry needs clear guidance on input parameter selection and justification. When conducting the final status survey used to verify that the dose assessment output has been properly implemented, guidance on the use of new survey technology needs to be developed. The MARSSIM approach is based on the concept of representative samples. Today we have technology that is capable of 100% survey coverage in some applications. Clearly this is superior to random samples but no guidance on how to use this approach is available. We should all agree that the better the guidance, the better the implementation and the greater the public acceptance.

The NRC is in the process of developing a supplement to the 1988 Generic Environmental Impact Statement (GEIS) on decommissioning. NEI commented in support of that effort. Clearly NRC-Research is involved with that effort. We concur that the supplement should include additional analysis of in-situ concrete rubble disposal and the entombment concept.

The analysis evaluating in-situ concrete rubble disposal should address realistic intruder scenarios. It should also explore realistic exhumation scenarios. In addition, isotopic migration through the concrete matrix needs to be addressed in a realistic way. These evaluations should be generic in nature and suitable for the GEIS supplement. The removal of concrete rubble containing trace residual activity and bringing in clean fill from off-site sounds appropriate until you assess the dose associated with the clean fill. In one recent assessment, the on-site dose would be significantly increased by this activity. A thorough GEIS would account for these factors.

The option known as entombment should also be addressed in the GEIS. It is prudent contingency management that the NRC develop a regulatory process that assures the public that reactors can be decommissioned safely even absent access to low-level radioactive waste disposal. The limited success achieved by the states over the past two decades in developing new disposal sites should make this point clear. To support this analysis, the significant work already done on engineered barriers, both here and abroad, should be focussed on this option. Questions such as what credit to give for passive water barriers, what are optimum designs to use, and what is the realistic duration of various engineered barriers, need to be answered.

In support of the material clearance rulemaking evaluation, significant work is needed to support a viable solution. This works scope includes a realistic inventory of materials for available for clearance, realistic industrial landfill disposal scenarios, and realistic material reuse scenarios. The focus on metal recycling has precluded the necessary focus on core material clearance issues. Failure to harmonization any future national standard with those of the international community and the resulting implications on international trade also need to be evaluated.

Decommissioning poses unique challenges in transportation. Large packages and high activity materials must be moved. Research support in two specific areas would be helpful in supporting the transportation of these materials. An activity based equivalent for the 1R/hr at 3 meters criteria which threshold for materials that require a Type B package, is needed. Absent and acceptable activity equivalent, direct dose-rate measurements and the resulting unnecessary personnel exposure will result. In addition, the development of an activity limit for LSA/SCO shipments that pose a comparable risk should be developed to support improvements in future IAEA safety standards.

Each of these identified issues contains opportunities to provide clarity, ease implementation, reduce unnecessary burden, while maintaining or improving safety.

NEI understands that the agency has plans to address many of the issues identified. As the ultimate user of the final products, the industry stands ready to assist your efforts to focus the limited resources available on research that is targeted to provide the greatest return on investment.

DEVELOPMENT OF PROBABILISTIC RESRAD COMPUTER CODES FOR NRC DECOMMISSIONING AND LICENSE TERMINATION APPLICATIONS.

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ABSTRACT

In 1999, the U.S. Nuclear Regulatory Commission (NRC) tasked Argonne National Laboratory to modify the existing RESRAD and RESRAD-BUILD codes to perform probabilistic, site-specific dose analysis for use with the NRC's Standard Review Plan for demonstrating compliance with the license termination rule. The RESRAD codes have been developed by Argonne to support the U.S. Department of Energy's (DOEs) cleanup of radioactively contaminated sites. Through more than a decade of application, the codes already have established a large user base in the nation and a rigorous QA support. The primary objectives of the NRC task are to: (1) extend the codes' capabilities to include probabilistic analysis, and (2) develop parameter distribution functions and perform probabilistic analysis with the codes. The new codes also contain user-friendly features specially designed with graphic-user interface. In October 2000, the revised RESRAD (version 6.0) and RESRAD-BUILD (version 3.0), together with the user's guide and relevant parameter information, have been developed and are made available to the general public via the Internet for use.

INTRODUCTION

On July 21, 1997, the U.S. Nuclear Regulatory Commission (NRC) published the License Termination Rule (Title 10, Code of Federal Regulations, Part 20 [10 CFR Part 20], Subpart E), which establishes requirements for nuclear facility licensees who are terminating their licensed operations. The NRC's approach to demonstrate compliance with the license termination rule is based on a philosophy of moving from simple, prudently conservative calculations toward more realistic simulations, as necessary, using dose modeling to evaluate exposure to residual radioactivity in soil and structures. Such potential exposures are evaluated for two general scenarios from residual contamination: building occupancy for building contamination, and site residential occupancy for soil contamination.

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The objective of dose modeling is to assess the total effective dose equivalent (TEDE) to an average member of the critical group* from residual contamination, including any contamination that has reached ground sources of drinking water. The assessment offers a reasonable translation of concentrations of residual radionuclide contamination (Bq/kg or pCi/g for volumetric contamination such as in soil and Bq/m^2 or dpm/100 cm^2 for surface contamination such as on building structure) into estimated radiation doses (mSv/yr or mrem/yr) to the public. Compliance with the NRC dose criteria can then be assessed by the modeling results. The assessment can be further used to obtain the derived concentration guideline levels (DCGLs) for cleanup activities. DCGL's are calculated (derived) residual levels or concentrations of radionuclides which corresponds to the NRC dose criteria by analysis of various release pathways and exposure scenarios such as direct radiation, inhalation, ingestion, etc. (NUREG-1575, MARSSSIM, [NRC 1997]).

As part of the development of site-specific implementation guidance supporting the License Termination Rule and development of a Standard Review Plan (SRP) on Decommissioning, the NRC recognized the need to perform probabilistic dose analysis with codes that could be used for site-specific modeling. Such modeling capabilities exist with the RESRAD (Yu et al. 1993) and RESRAD-BUILD (Yu et al. 1994) codes (Figure 1). The RESRAD and RESRAD-BUILD computer codes have been developed by Argonne under U.S. Department of Energy (DOE) sponsorship for use in evaluating radioactively contaminated sites and structures, respectively. Both are widely used in cleanup operations in the United States and abroad. These DOE codes possess the following attributes: (1) the software has been widely accepted and there is already a large user base, (2) the models in the software were designed for and have been successfully applied at sites with relatively complex physical and contamination conditions, and (3) verification and validation of the codes are well documented (Cheng et al. 1995; Gnanapragasam and Yu 1997a, 1997b; NUREG/CP-0163 [NRC] 1998). The RESRAD codes have been used primarily to derive site-specific cleanup DCGLs based on the deterministic method.

In 1999, the U.S. Nuclear Regulatory Commission (NRC) tasked Argonne National Laboratory (Argonne) with adapting the existing RESRAD and RESRAD-BUILD codes for use in site-specific dose modeling in accordance with NRC guidance in the Standard Review Plan (SRP) for Decommissioning for demonstrating compliance with the license termination rule. For NRC's intended use, the codes are being revised to be consistent with the current NRC guidance for dose modeling being developed in the SRP for Decommissioning. Thus, the primary objectives of Argonne's effort are to (1) extend the codes' capabilities to include probabilistic analysis, and (2) develop parameter distribution functions and perform probabilistic analysis with the codes. The two codes incorporate pathway analysis models designed to evaluate the potential radiological dose to an average individual of the critical group who lives or works at a site or in a structure contaminated with residual radioactive materials.

*The critical group is defined as an individual or relatively homogenous group of individuals expected to receive the highest exposure under the assumptions of the particular scenario considered (NUREG/CR-5512, Vol. 1, 1992). The average member of the critical group is an individual assumed to represent the most likely exposure situation.

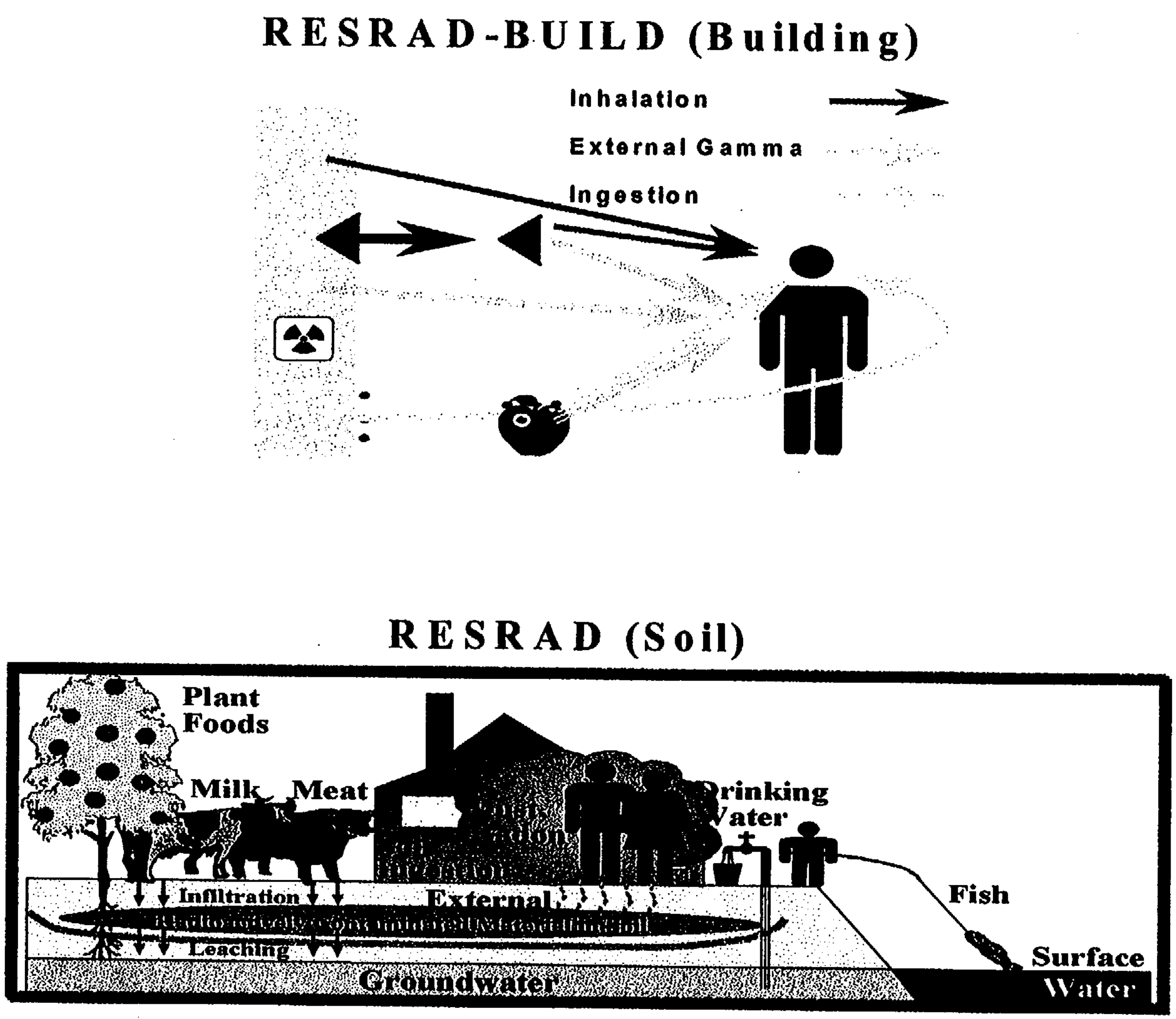

Figure 1. Schematic Illustration of RESRAD (Soil) and RESRAD-BUILD (Building) Pathway Analysis Approach

In meeting the NRC's objectives, the enhanced RESRAD codes are designed to comprise three major capabilities: probabilistic sampling, analytical capabilities, and user-friendly interface. The probabilistic sampling module incorporates the Latin Hypercube sampling (LHS) method (Iman et al. 1984) to perform random sampling of input parameters. The deterministic RESRAD codes serve as the core analytical modules for dose calculation. The codes are further equipped with user-friendly input and output interface features.

The code development effort is supported by a rather extensive data search and development. To this end, a total of nearly 200 input parameters for RESRAD and RESRAD-BUILD combined were screened and ranked for their relative importance to dose analysis. About one third of the parameters were identified as important with detailed probability distribution functions developed. Such data distribution was developed using a rather extensive data search based on national variability followed by an analysis of the distribution function.

The software was designed with a user-centered approach. It provides easy access of output through interactive tabular windows; interactive graphical windows; fixed tabular reports; and a complete, formatted database. The output results were chosen to support resultant dose distribution statistics, distributions, and correlations with the input variables. These results can be queried on the basis of time since contaminant placement, initial radionuclide concentrations, environmental transport and exposure pathways. The calculation also identifies the peak doses over the specified time (within 1,000 years per the NRC's SRP guidance) for the purpose of demonstrating compliance with the license termination rule.

The integrated software package leverages the user's familiarity with standard Microsoft Windows™ tools and the family of RESRAD software tools. The probabilistic screens are tightly integrated with the previously identified default distributions for the input variables. Additionally, users also have a variety of options to enter site-specific parameters through the newly developed features. The software offers feedback to quickly identify the default and site-specific distributions. The user can also graphically preview the distribution shape. The integrated RESRAD and RESRAD-BUILD codes are designed to operate on Microsoft Windows™ 95, 98, 2000, and NT platforms.

APPROACH

The objective of dose modeling is to assess the total effective dose equivalent (TEDE) to an average member of the critical group from residual contamination, including any contamination that has reached ground sources of drinking water. The assessment offers a reasonable translation of residual contamination into estimated radiation doses to the public. The estimated doses can then be compared with the NRC 10 CFR Part 20 dose criteria for decommissioning to evaluate compliance with the license termination rule.

The task of developing probabilistic RESRAD and RESRAD-BUILD codes is discussed in subsequent sections. The effort represents three major areas of development: (1) identify key parameters and develop parameter distributions (i.e., parameter categorization, parameter ranking, and development of parameter distributions), (2) develop probabilistic modules, and (3) perform code testing and analysis (i.e., probabilistic analysis and testing of probabilistic modules). Because of the interrelationships among these efforts, an iterative process is used to attain optimal results. For instance, preliminary code testing was frequently conducted to test the code at the various stages of development. Likewise, periodic testing of the preliminary versions of the codes also provided valuable feedback to the development of parameter distributions.

The strategy to the approach was to fully use the capabilities of the existing RESRAD codes and provide needed enhancements for satisfying the NRC's Standard Review Plan objectives for site-specific dose analysis (Figure 2). In the process of development, consistencies were also maintained, to the extent possible, with the generic methodologies described in NRC publications (NRC 2000; NUREG/CR-1549, 1998; NUREG/CR-5512, Vols. 1-3, 1992, 1998, 1999).

PARAMETER AND DATA DEVELOPMENT

Parameter Categorization. The first step was to list and categorize the total combined input parameters (about 200) used in the RESRAD and RESRAD-BUILD codes. The parameters were first listed into three major categories: physical, behavioral, and metabolic. Parameters that would not vary with the conditions of receptors were classified as physical parameters. Examples include soil leaching factor and contamination thickness. Parameters that exhibit a relationship with the receptor's behavior and the scenario definition were classified as behavioral parameters. Examples include building occupancy factor and food ingestion rate. Parameters that represent the metabolic characteristics of the potential receptor and that would be independent of the scenario being considered were classified as metabolic parameters. Examples include dose conversion factors. Certain parameters, such as inhalation rate (both behavioral and metabolic), have been found to exhibit a dual characteristics. In those instances, a dominant characteristic was assigned to the parameter.

Of a total of 145 RESRAD parameters (discounting flag parameters), 107 have been identified as physical (P), 25 behavioral (B), and 10 metabolic (M). For a total of 50 parameters of RESRAD-BUILD, 33 were identified as physical, 12 behavioral, and 5 metabolic (Kamboj et al. 1999).

Parameter Ranking. A strategy was developed to rank the RESRAD and RESRAD-BUILD input parameters and identify parameters for detailed distribution analysis. Depending on their importance, parameter distributions were characterized as high, medium, or low priority. The parameters were ranked on the basis of four criteria: (1) relevance of the parameter in dose calculations, (2) variability of the radiation dose as a result of changes in the parameter value, (3) parameter type (physical, behavioral, or metabolic), and (4) availability of data on the parameter in the literature. A composite scoring system was

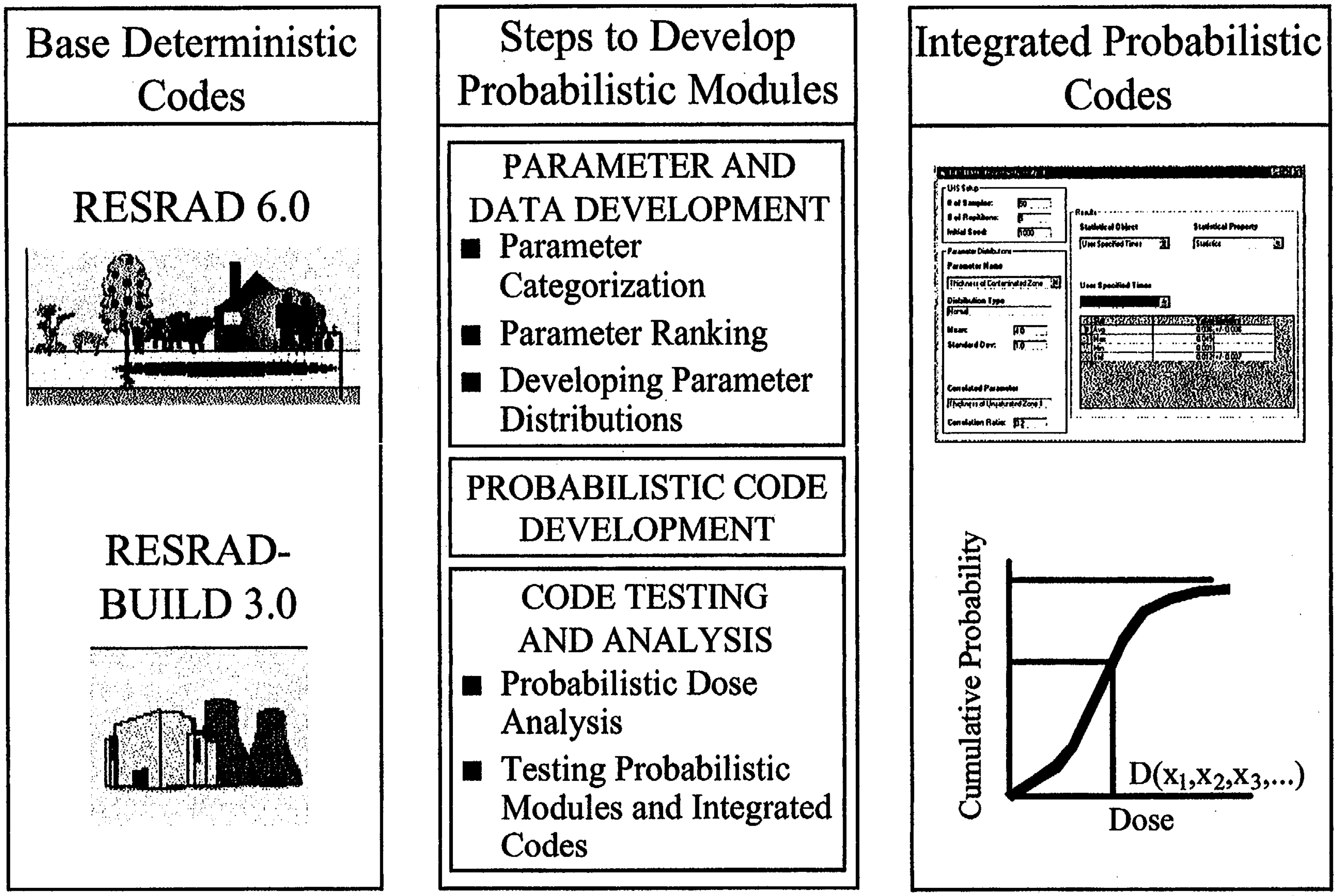

Figure 2. Approach to the Development of Probabilistic RESRAD and RESRAD-BUILD Codes

developed to rank the parameters based on each criterion, with a low score assigned to parameters with a higher priority and a high score assigned to parameters with lower priority under the considered criterion. In all, parameters for RESRAD are ranked as follows: 10 as high priority (H), 39 for medium priority (M), and 96 as low priority (L). For RESRAD-BUILD, there are 4 ranked as high priority, 20 as medium priority, and 24 as low priority (Cheng et al. 1999).

Parameter Distribution. Parameter distributions were developed for those identified as high priority and for the majority of the medium priority in the RESRAD and RESRAD-BUILD codes. A total of 70 parameters were selected for analysis (discounting flag parameters, 66 were assigned a distribution) . These parameters were deemed to be the ones most relevant to the NRC objective of dose analysis for demonstrating compliance with the radiological criteria stipulated in the decommissioning and license termination rule. Development of parameters distributions entailed the following basis and assumptions: (1) use nationwide data representation, (2) use of the most relevant and up-to-date data sources across the nation, (3) obtain the best fit to characterize the distribution, and (4) assume an average adult male as a receptor (constrained by current availability of dose conversion factors). Compilation of the data entailed an extensive literature search using library and Internet resources. The focus was placed on analyzing the available data and making the most plausible distribution assignments for each selected parameter (Biwer et al., 2000). In general, parameters were characterized into five distribution types: uniform/log uniform, triangular, normal, log normal, and empirical.

In this process, it was recognized that many of the national parameters in question may not be well suited for site-specific applications, since they can vary significantly from site to site or even within the same site. Nevertheless, the users are encouraged to develop site-specific data distributions where warranted using the similar methodology for deriving the appropriate representation for a particular parameter distribution. It is also recognized that the derived distribution information contains varying quality due to the availability and quality of the original data sources. In general, data quality for RESRAD parameters tends to be better than that of RESRAD-BUILD parameters. The primary reason is attributable to the much longer history of dealing with site (i.e. soil) cleanup than that of the building decontamination and decommissioning. One example is the surface contaminant emission rate, which tends to dominate the estimated dose from a contaminated building surface (using RESRAD-BUILD). Existing information on this particular parameter is, however, rather scant. Continued research on such key parameters is certainly warranted and strongly recommended.

PROBABILISTIC MODULE DEVELOPMENT

Probabilistic Modules. The RESRAD and RESRAD-BUILD computer codes have been developed by Argonne under sponsorship of the U.S. Department of Energy (DOE) for use in evaluating, by a deterministic approach, the radioactively contaminated sites and structures, respectively. Both are widely used in cleanup operations in the United States and abroad. Both codes are pathway analysis models designed to evaluate the potential radiological dose to an average individual of the critical group who lives or works at a site or in a structure contaminated with residual radioactive materials.

As part of the ongoing effort to meet NRC's objectives, external modules equipped with probabilistic sampling and analytical capabilities were developed for RESRAD and RESRAD-BUILD. To this end, the major aspects of the software system design include: (1) development and integration of the existing deterministic RESRAD codes with the external Latin Hypercube sampling (LHS) software (Iman et al. 1984), (2) incorporation of parameter distribution data previously discussed, (3) development of input and output interfaces of the integrated system, (4) development of testing methods, and (5) incorporation of software quality assurance (QA) methods (LePoire et al. 2000). Design of the integrated software system is shown in Figure 3.

The modules are further equipped with user-friendly input and output interface features to accommodate numerous parameter distribution functions and result display requirements. The integrated system, consisting of the codes and the interface modules, is designed to operate on Microsoft Windows™ 95, 98, and NT platforms.

CODE TESTING AND ANALYSIS

Probabilistic Module Testing. The testing effort was performed periodically to satisfy the project QA requirements and to ensure consistency of development throughout the process. Testing consists of four major components:

- Testing of the probabilistic input data sampling program. This effort is to ensure the compatibility and functionality of the externally acquired Latin Hypercube sampling (LHS) program prior to its adaptation to the system of RESRAD codes.

- Calculation integration testing. The probabilistic modules were tested during the development mode for their proper execution and for the reasonableness of the results. Testing was conducted on radionuclides, pathways, input correlations, and comparison with deterministic results.

- Output interface testing. Testing focused on the output interface that is relevant to the accuracy of the results. This includes a comparison of percentile and statistics of interactive tables and reports. Results reported in the tables and the corresponding graphic outputs were also compared for consistency. Results on peak dose were also examined.

- Integrated testing. The calculation, interface, and distribution aspects of the fully integrated system were tested. A scenario case was used and the results from the software are interpreted. The interface was reviewed with modern user interface heuristics as a guide. The distribution process was checked for completeness,

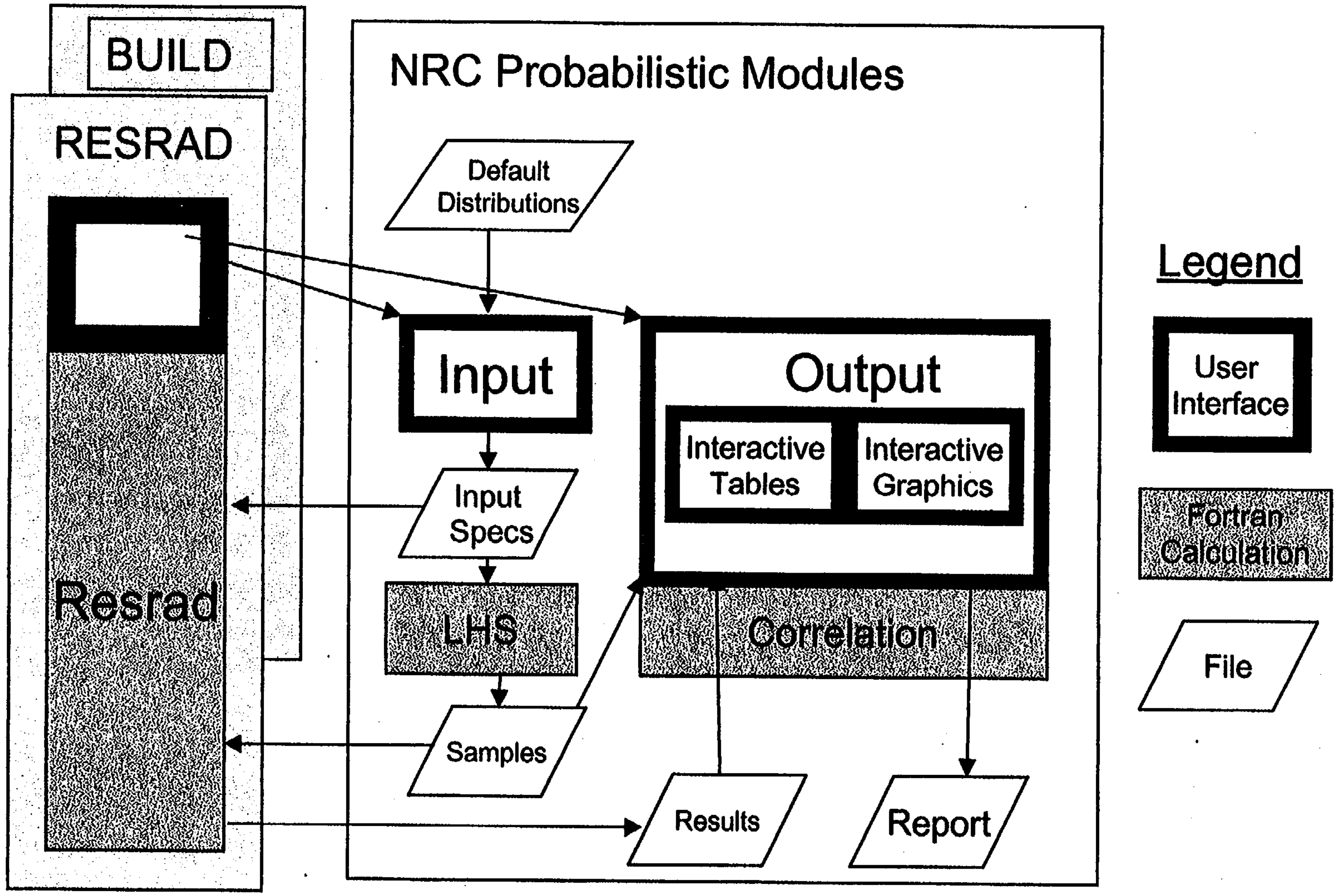

Figure 3. Design of an Integrated Software System for the Probabilistic RESRAD Codes

compatibility, and security from viruses over a range of operating systems. Testing was also conducted independently by the NRC staff as well as the industry. Feedback was incorporated into the final code development.

Testing procedure and results of the software development were documented by LePoire et al. (2000) and were found to satisfy the QA requirements. As is the case with any newly released codes, further testing can be realized as the codes are being released for use by the industry as well as the interested public.

Probabilistic Dose Analysis. The effects of parameter distribution on the estimated doses, taking into account parameter correlations, for the residential scenario in RESRAD and for the building occupancy scenario in RESRAD-BUILD were assessed. The analysis took into account long-term transport of residual radionuclides in the environment and associated exposure pathways. For RESRAD, the peak dose within a 1,000-year time frame was captured, and for RESRAD-BUILD, the initial dose (i.e., at time 0) was calculated and used as the peak dose. The probabilistic analysis was performed by using the stratified sampling of the Latin hypercube sampling method for a collection of input parameter distributions. Figure 4 shows an example of the analysis using RESRAD on a residential scenario. The results illustrate the variability of the output dose results with contamination setting for a few given radionuclides. Given a specific regulatory interpretation in terms of cumulative probability in dose, the results render a clear method for demonstrating compliance. The probabilistic analysis has demonstrated the process of using the integrated RESRAD and RESRAD-BUILD codes and the probabilistic modules, together with the parameter distributions, for dose assessment at a relatively complex site (Kamboj et al. 2000).

SUMMARY AND CONCLUSIONS

The advanced versions of RESRAD (version 6.0) and RESRAD-BUILD (version 3.0) have been developed out of the existing deterministic versions to support NRC's Standard Review Plan for its license termination activities. The new codes are equipped with probabilistic analytical capabilities for radiological dose analysis in site-specific analysis.

The development took advantage of the codes' wide circulation and popular use in environmental cleanup activities throughout the nation and abroad. An external module containing the advanced Latin Hypercube sampling technique was incorporated into the codes. To support the analysis, a database containing parameter distribution was also developed. This was accomplished by identifying key parameters from a total of nearly 200 parameters combined for both codes. Best available data were compiled for those key parameters, taking into account the national variability, and fitted into the most representative statistical representation. The codes were equipped with input/output features designed to offer users a high degree of user friendliness. Outputs are presented in tabular as well as graphic forms by radionuclide or pathway. Peak doses over time are identified and compiled for use in evaluating compliance with the license termination rule.

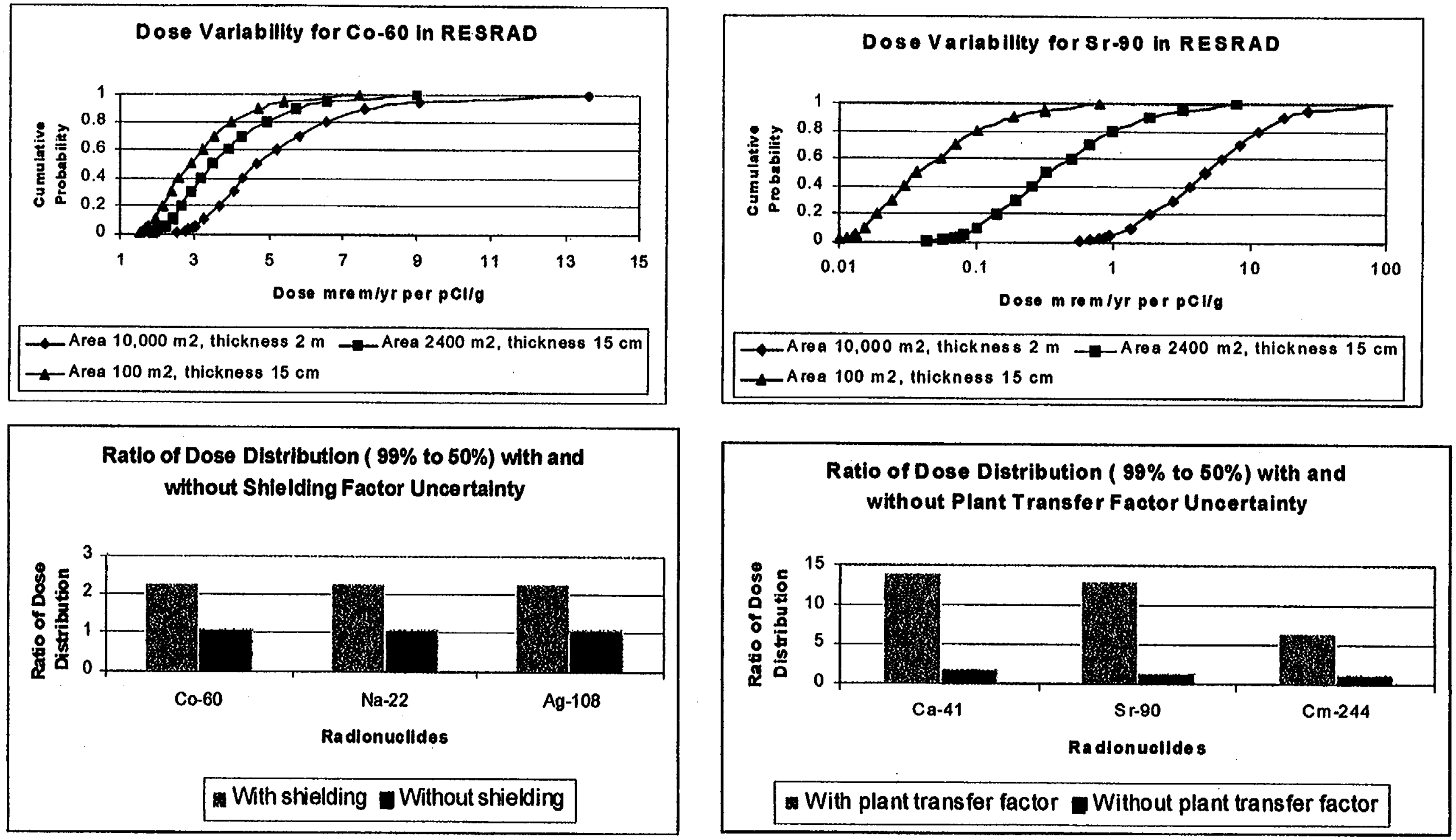

Figure 4. An Example of Dose Analysis Using the Probabilistic RESRAD Code for Residential Scenario

Extensive testing has been conducted according to the project QA plan prior to the release of the codes. The integrated RESRAD and RESRAD-BUILD codes are designed to operate on Microsoft Windows™ 95, 98, 2000, and NT platforms. The codes and the user's guide are currently available for download at the NRC's website.

The development of the codes also provides future research opportunities. These potentials areas may include (1) a full-scale sensitivity analysis for analyzing the parameters to identify areas of research, (2) performing case studies to verify the application of the probabilistic method (as opposed to the deterministic approach), (3) extending the approach to other areas of analysis regarding pathway dose and risk analysis, and (4) establishing an interagency effort in maintaining consistencies in model development and application protocols.

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Surveying for Radionuclides in Inaccessible or Complex Geometry Materials

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Abstract

Clearance surveys are performed on materials for recycle or reuse to demonstrate that appropriate release criteria have been met. Conventional clearance survey activities include scanning, direct measurements of surface activity, smear sampling and other miscellaneous media sampling. A practical challenge that arises in the performance of clearance surveys is surveying for radionuclides that are present in inaccessible or complex geometry materials. The proposed research is to develop survey and analytical methods to measure residual radioactivity in inaccessible areas to assure that applicable criteria have been met. Specific conditions and limitations related to inaccessible contamination measurements are identified. Special emphasis will be given for mixtures of clearance materials, such as buried or embedded piping systems and scrap metal piles. Proposed research activities to address these inaccessible areas are provided in this paper.

Introduction

The clearance of materials is a common activity at operational sites as well as those undergoing decommissioning. The clearance objective is to release materials (e.g., items and equipment) for unrestricted use—either for recycle or reuse—via a comprehensive radiological survey. The clearance survey process typically includes a nominal 100% survey of all the material being released for unrestricted use—which usually meant that the surveyor would scan as close to 100% of the accessible material surfaces as possible. In addition to surface scans, survey activities included direct measurements of surface activity and smears for assessing removable activity levels.

A question that often comes up is how to handle the release of materials that have inaccessible areas that may be contaminated. Obviously, if the material surfaces are inaccessible, then by definition, it is not possible to demonstrate that release criteria have been satisfied using conventional survey activities. In this case, a couple of options exist. First, the material might not be released for unrestricted use—that is, it might be concluded that since surfaces are not accessible, they will be assumed to be contaminated at levels greater than the release criteria. Thus, the materials might be disposed of as radioactive waste. In fact, this has been the historical approach to dealing with materials that have inaccessible surfaces.

A second alternative might be to make the surfaces accessible either by cutting or dismantling the material, or by using specialized survey equipment (e.g., small detectors) to make the surfaces accessible. This option requires the use of additional resources beyond that required for conventional clearance surveys. This paper addresses a number of research opportunities for handling materials that have inaccessible areas.

Inaccessible Material Scenarios

It is important to recognize the various inaccessible material scenarios that can occur during the clearance of materials. Perhaps the most common scenario is when the interior surfaces of scrap equipment, such a pumps, motors, and other equipment, are contaminated. These items can be contaminated through a number of mechanisms, including their operation in airborne contamination areas where air is drawn into the equipment, contaminating internal surfaces. Similarly, lubricating oil can become contaminated which further contaminates a number of components within the scrap equipment. Thus, due to the small openings on these items, conventional survey activities to address the potential for internal contamination is nearly impossible.

Another inaccessible material scenario is interior surfaces of pipes that are difficult to access—such as buried or embedded pipes. Buried and embedded pipes may become contaminated by virtue of their function of transporting radioactive liquids or gases. Buried pipes are usually at some depth beneath the soil surface and cannot be accessed unless they are excavated. Process piping, such as that associated with nuclear power reactor systems, can be embedded in concrete. Typically, the small diameter of embedded piping makes it extremely difficult to access their interior surfaces.

One final inaccessible material scenario includes some of the material surfaces in a scrap metal (or other material) pile. This complex geometry is somewhat different from the first two scenarios in that these surfaces can be made accessible, but the effort required to separate the materials for survey might be considered too labor-intensive to warrant conventional clearance surveys. Therefore, it might be worthwhile to consider releasing a pile of scrap metal by taking *in situ* gamma spectrometry measurements of the scrap metal pile. In this case, some of the scrap metal surfaces are considered inaccessible because they do not directly contribute to the detector's response. However, provided that a sufficient fraction of gamma radiation from the contamination is detected, *in situ* gamma spectrometry might provide a reasonable clearance technique for scrap metal piles.

Make Inaccessible Area Accessible

One strategy that can be considered when dealing with materials that have inaccessible areas is to make the inaccessible areas accessible. For example, this can be accomplished by dismantling scrap equipment or by excavating the buried or embedded pipes. Inaccessible areas that might require disassembly include small pumps, motors, hand tools, power tools, and electrical control panels. These materials are assumed to require some amount of disassembly to provide access to their interior surfaces. The dismantlement activities might be deliberate to ensure that the item is still functional following the efforts to gain access to internal surfaces. Conversely, cutting techniques can be employed to expedite the process if reuse is not an option.

Another technique that may be considered is the use of thermoluminescent dosimeters (TLDs) or small detectors to measure surface activity levels within buried and embedded piping systems. TLDs can be deployed for some period of time within small bore piping or conduit to respond to the contamination levels on these interior surfaces. An important aspect of this application is the calibration of the TLDs to surface activity in these pipe geometries. Small detectors such as miniature GM detectors and other "pipe-crawling" detector systems have been used to assess surface contamination in pipe systems.

Empirical Studies using *In Situ* Gamma Spectrometry

As mentioned previously, it may be possible to release scrap metal by taking *in situ* gamma spectrometry measurements of the scrap metal pile. The proposed research would consist of fabricating small sources of radioactivity that would be placed at various locations throughout the scrap metal pile. These empirical studies would evaluate the levels of contamination that are detected for a range of gamma-emitting radionuclides—e.g., Co-60, Cs-137 and Am-241—using the *in situ* gamma spectrometer. The proposed research would determine the optimal *in situ* gamma spectrometer measurement locations and use of detector collimation. Thus, *in situ* gamma spectrometer measurements for scrap metal in a complex geometry that renders some of the surfaces inaccessible may be a viable release survey option.

Modeling/Simulations for Buried Pipe

Another proposed area of research is to assess the level of surface contamination within a buried pipe by making gamma measurements outside of the buried pipe at depth. That is, boreholes would be installed adjacent to the buried pipe and gamma radiation measurements performed with a NaI scintillation detector to infer the surface activity within the pipe. The proposed research design would consist of installing a series of PVC pipes perpendicular to a buried pipe that contains a known amount of gamma-emitting surface activity. Gamma radiation measurements would be performed using NaI detectors located within the PVC pipes. The PVC pipes would be open at ground level and positioned approximately 3 to 5 cm from the buried pipe.

MicroShield™ modeling would be used convert surface activity within the buried pipe to exposure rates at the NaI detector measurement locations, and the relationship of exposure rate to count rate would be determined for the NaI scintillation detector. That is, a calibration factor for the NaI scintillation detector would be determined to allow the conversion of detector counts per minute (cpm) to exposure rate (μR/h). The MicroShield™ modeling results would be benchmarked against the results obtained from the known activity level within the buried pipe. The model accounts for the radionuclide source strength, geometry, and attenuation produced by the pipes and intervening soil thickness. Therefore, gamma radiation measurements performed in the PVC pipes are converted to exposure rate measurements, and are used to estimate the activity in the buried pipe for an assumed activity distribution in the buried pipe.

REGULATORY EFFECTIVENESS: WHAT IT IS AND WHAT IT SHOWS FOR THE STATION BLACKOUT RULE

By Bill Raughley
U.S. Nuclear Regulatory Commission

BACKGROUND

- **REGULATORY EFFECTIVENESS IS AN NRC PERFORMANCE GOAL**

  — "Make NRC Activities and decisions more effective, efficient, and realistic"

  -Maintaining quality of technical base
  -Applying Principals of Good Regulation
  -Using realistic information
  -Improving predictability and consistency of decisions

  — Commission interest in effectiveness of SBO and ATWS rules

- **SBO RULE**

  — SBO definition and risks

  — 10 CFR50.63, "Loss of all alternating current power" risks, coping, AAC

  — Historical highlights:

  -Regulations before the SBO rule
  -Generic Safety Issue GSI A-44
  -Technical basis(NUREG-1032), regulatory analysis(NUREG-1109)

  — SBO regulatory document s

  -SBO Regulatory Guide RG 1.155 (NSAC-108 & NUMARC 87-00, Rev 0)
  -Maintenance Rule RG 1.160 (NUMARC 93-01)
  -Inspection documents

SBO ASSESSMENT

- "FINAL REPORT - REGULATORY EFFECTIVENESS OF THE STATION BLACKOUT RULE" (ADAMS ML0037420860)

- REGULATORY EFFECTIVENESS
  - A regulation is effective if its expectations (desired outcomes) are being achieved
  - Regulation includes the rule, regulatory guide, and inspection documents

- SCOPE
  - Is the SBO rule effective and if any areas need attention
  - GSI-23 on RCP seal failure previously addressed

- METHOD
  - Compare regulatory expectations to outcomes using objective measures in the areas of risk, coping time, EDG reliability, and value-impact
  - Used operating experience to be realistic
  - Solicited internal and public comment on reasonableness of method, appropriateness of conclusions, and future assessment topics

SBO RULE EXPECTATIONS

- ♦ MEAN INDUSTRY RISK REDUCTION PER PLANT OF 2.6E-05/RY
- ♦ MINIMUM SBO COPING CAPABILITY
  - — 2-, 4-, 8-, or 16-hours
  - — 100 plants would develop procedures and training to cope
  - — 39 plants would make modifications to achieve desired coping capability
- ♦ INDIVIDUAL EDG TARGET RELIABILITY LEVELS OF 0.95 OR 0.975
  - — Assumes maintenance and test out of service (MOOS) assumed to be small (0.007) to achieve failure rates
  - — Different boundaries than technical basis and different guidance for data to use in counting failures
  - — Maintenance rule regulatory guide options to monitor EDG performance
  - — SBO and maintenance rule inspection procedures use NUMARC trigger values to assess RG 1.155 compliance
- ♦ VALUE-IMPACT RATIO (PERSON-REM AVERTED/$MILLION) OF 2400, RANGE OF 700-5000

RESULTS

- **MEAN INDUSTRY WIDE RISK REDUCTION OF 3.2E-05/RY**
  - Plants that had the greatest vulnerability did the most
  - Shutdown with power supply unavailability may increase risk

- **SBO COPING CAPABILITY**
  - All plants have 4- or 8- hour coping capability, procedures, and training
  - 72 plants made modifications

- **EDG RELIABILITY-BASED ON UNIT AVERAGE EDG SAFETY PERFORMANCE**
  - Generally better than 0.95 with and without MOOS
  - 0.975 difficult to achieve due to MOOS while reactor is running may erode risk benefits gained from SBO rule implementation
  - Inconsistent use of reliability terms
  - Additional risk reduction possible for some licensees

- **VALUE-IMPACT RATIO OF 954**
  - Value better due to greater than expected risk reduction
  - 19 additional power supplies contributed to greater than expected impact and added value due to operating flexibility

The Number of Plant Units in Station Blackout Core Damage Frequency Ranges Before and After Station Blackout Rule Implementation

Parameter	Number of Plants in SBO CDF Range (E-05 per reactor-year)											
SBO CDF Range	< 0.5	0.5 .99	1.0 1.49	1.5 1.99	2.0 2.49	2.5 2.99	3.0 3.49	3.5 3.99	4.0 4.49	4.5 4.99	5.0 9.99	10 35
Before SBO rule Implementation	5	13	14	7	13	4	9	5	4	3	13	10
Expected After SBO rule Implementation	23	23	14	9	6	5	6	5	4	0	5	0
Actual Outcome After SBO rule Implementation	46	22	13	17	1	3	1	3	0	1	1	0

Probabilistic Risk Assessment/ Individual Plant Examination Sensitivity Analyses

Description of Modification	Effect on Overall Risk (Percent Reduction of Plant CDF)
Adding EDGs	
Calvert Cliffs (one safety and one nonsafety EDG)	24
Turkey Point (two safety EDGs)	20
Adding safety EDG	
Diablo Canyon	14–18
Add nonsafety EDG for site	
Arkansas Nuclear 1	23–36
Arkansas Nuclear 2	43–47
Procedural	
Arkansas Nuclear 1: EDG service water supply valve open	7
Monticello: Battery load shed	17
AC cross-tie	38
Extend battery life from 2 to 4 hours	
Arkansas Nuclear 1	16

Effects of Maintenance Out of Service
While the Reactor is at Power on Emergency Diesel Generator Reliability

EDG target reliability	Mean industry unit average EDG reliability		
	without MOOS	with MOOS	decrease in reliability
0.95	0.985	0.954	0.034
0.975	0.978	0.967	0.012

INSIGHTS FROM OPERATING EXPERIENCE

- MODIFICATIONS DUE TO THE SBO RULE HAVE BEEN USED TO PROVIDE FOR SAFE SHUTDOWN AND ECONOMIC BENEFIT

- PROVIDE ADDITIONAL DEFENSE IN DEPTH FROM CHANGING OFFSITE POWER TRENDS DUE TO DEREGULATION

- POTENTIAL ALTERNATE AC POWER SOURCE UNAVAILABILITY

CONCLUSIONS

- SBO RULE WAS GENERALLY EFFECTIVE AND THE COSTS WERE REASONABLE.

- CONSISTENT WITH PRINCIPLES OF GOOD REGULATION THAT INCLUDE CLARITY AND RELIABILITY THERE ARE OPPORTUNITIES TO REVISE SELECTED REGULATORY DOCUMENTS
  - Establish common reliability terms, measurements, criteria in the regulatory guidance
  - Practical and risk informed shutdown guidance with power supply unavailability
  - Inspection document revisions

- LESSONS LEARNED
  - To the extent that the regulations are revised should:
    - -Ensure consistent interpretation and use of terms, measurements, and criteria
    - -Include measurable objectives to facilitate evaluation of its regulatory effectiveness

Unnecessary Regulatory Burden

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Licensing Director,

Zion/Dresden Unit 1 Decommissioning

October 25, 2000

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Introduction

- **Identification of unnecessary regulatory burden**
- **Meeting with NRC Office of Research**
- **Proposed reductions identified in these areas:**
  - » ***radiation protection***
  - » ***fitness for duty***
  - » ***physical protection***
  - » ***emergency preparedness***
  - » ***nuclear fuels***
  - » ***reporting requirements***

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Radiation Protection

- Advise workers of their dose only upon request or if >100mrem/yr
- Label individual containers of licensed material in radiologically posted area (RPA) only when container's dose rate/contamination level is > ambient for RPA\
- Attempt to obtain records of cumulative occupational dose only for planned special exposure
- Eliminate annual report of personnel receiving > 100 mrem exposure

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Fitness for Duty

- Reduce best-effort verification of employment history of individuals to 3 years
- Eliminate the fitness-for-duty permanent record book
- Eliminate the requirement to test for drugs on a for-cause alcohol test
- Increase opiate metabolite cut-off level to 2000 ng/ml for initial and confirmatory tests
- Eliminate quality control specimens of test samples

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Physical Protection

- Eliminate requirement to protect against insider threat
- Only classify defensive plans and number of armed responders as Safeguards Information
- Eliminate 0.2 ft.-candle lighting requirement from all other areas within the protected area (PA) beside the isolation zone
- Eliminate the requirement for vehicles entering the PA to be escorted by a member of the security organization

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Physical Protection

- **Eliminate NRC as the intermediary in the criminal history check**

- **Eliminate requirement to perform credit, education, and military service checks as part of background checks**

- **Eliminate vital area and equipment designation**

Emergency Preparedness

- **Reduce plume exposure pathway emergency planning zone to five mile radius**
- **Eliminate requirement to update evacuation time estimates**

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Nuclear Fuels

- **Relocate the Technical Specifications (TS) value of the minimum critical power ratio safety limit to a licensee controlled document**

- **Relocate the methodologies referenced in the TS used to determine the limits in the core operating limits report (COLR) to the COLR**

- **Eliminate reporting requirement regarding changes in peak cladding temperature**

- **Use more realistic approach for calculating heat generation rates from radioactive decay of fission products**

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Reporting Requirements

- **Eliminate requirement to submit fitness-for-duty program performance data every six months**

- **Eliminate requirement to include 10 copies of updated final safety analysis report (UFSAR) replacement pages**

- **Eliminate requirement to submit annual radioactive effluent report**

- **Eliminate requirement to submit changes to physical security plans made without prior NRC approval**

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Reporting Requirements

- **Eliminate requirement to submit changes to emergency plan or implementing procedures made without prior NRC approval**

- **Eliminate requirement to submit annual property insurance coverage report**

- **Eliminate requirement to submit irradiated fuel management and funding plan**

- **Eliminate requirement to submit periodic summary reports of safety evaluations**

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Reporting Requirements

- Eliminate requirement to submit annual financial report
- Eliminate requirement to submit annual report of radioactive effluents for dry cask storage types which do not have effluents
- Eliminate requirement to annually submit evidence of guarantee of payment of deferred insurance premiums

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Summary

- **Approximately 30 items identified to reduce unnecessary regulatory burden**
  - » **bases for proposed reductions do not need further research**
  - » **proposed reductions represent > $4M annual savings for ComEd**

- **NRC should pursue direct final rulemakings for these simple regulatory burden reductions**

HIGH-LEVEL GUIDELINES
FOR
PERFORMANCE-BASED ACTIVITIES

PRESENTATION TO WATER REACTOR SAFETY MEETING

OCTOBER 25, 2000

N. Prasad Kadambi, RES/REAHFB

Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission

Outline

- Overview

- Case Studies Involving Application of the Guidelines

  - Case Study of Hypothetical Regulatory Framework for "Control of Combustible Gases"

  - Case Study on Subpart H to 10 Cfr Part 20, "Respiratory Protection and Controls to Restrict Internal Exposure in Restricted Areas"

- Interrelationship among Regulatory Initiatives

- Conclusion

Overview

- Commission Paper Secy-00-191 (Available on Nrc Website) Provides Final Guidelines, Responses to Public Comments and Case Studies.
- Guidelines Were Developed with Extensive Internal and External Stakeholder Input.
  - Considerable Support for Performance-based Approaches Evident
  - Concern Has Been Expressed That Focus May Be Too Much on Reducing Regulatory Burden
  - Responses to Public Comments Focuses on Published Questions
- Two Case Studies Performed to Test Whether the Guidelines Are Useful.
- Staff Concluded That Guidelines Are Ready for Agency-wide Application
- Guidelines and Process for Applying Them Expected to Evolve and Improve with Experience.

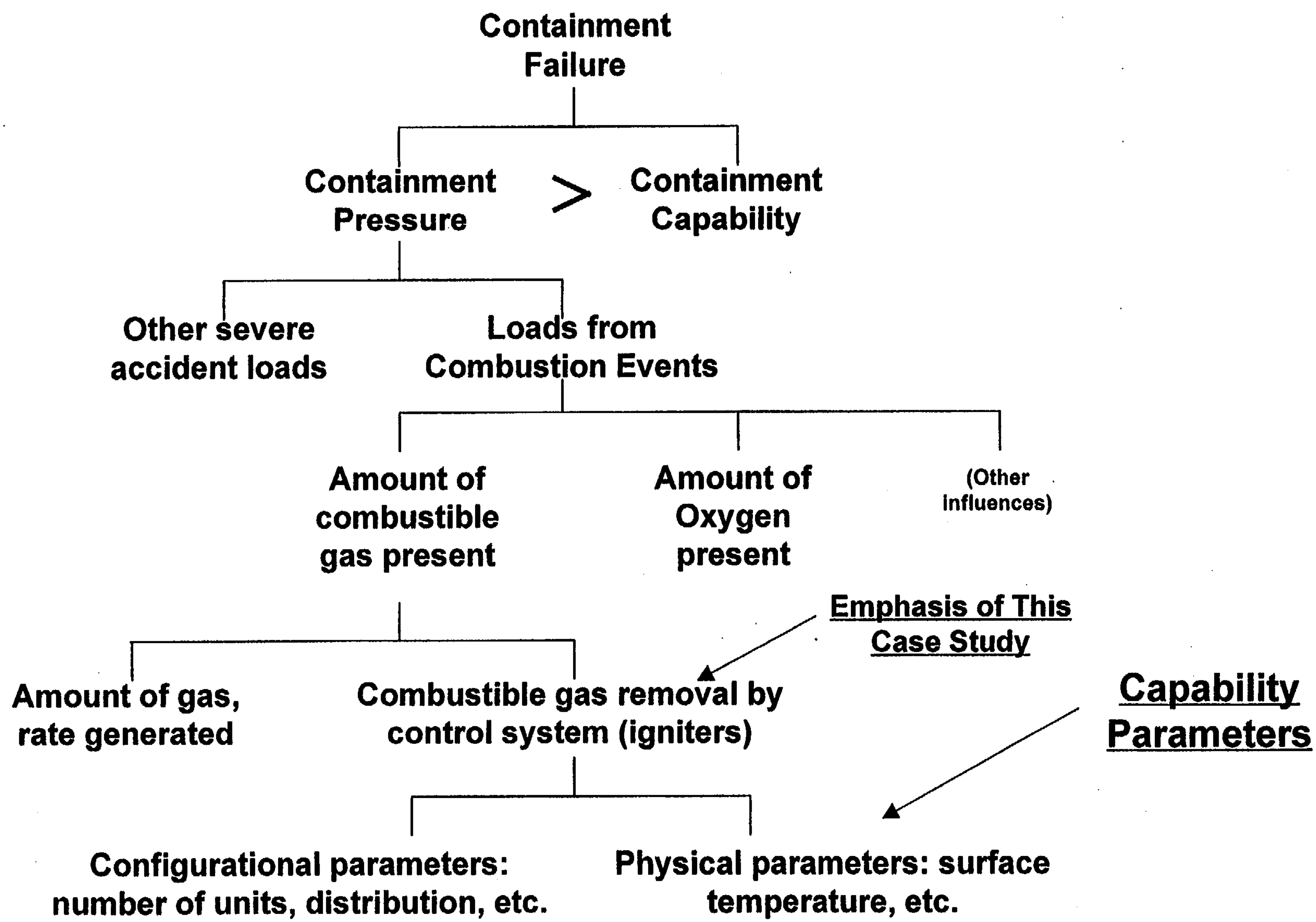
Overview of High-level Guidelines

- Guidelines Based on Commission's High-level Concepts and Direction
- Guidelines Provide Structure and Process for Relevant Definitions in White Paper on "Risk-informed & Performance-based Regulation"
- Three Groupings of Guidelines:
    - Guidelines to Assess Viability
    - Guidelines to Assess Change
    - Guidelines to Assure Consistency with Other Regulatory Principles
- Risk Information Used For:
    - Establishing Level of Performance
    - Metrics, Thresholds And/or Regulatory Response
    - Note Lack of Quantitative Risk Models

Case Study 1
Combustible Gas Control

- This case study applies the viability guidelines to a hypothetical regulatory framework on combustible gas control in certain containment types
- Risk information can be used to establish what the requirement needs to accomplish:
  - Safety mission (what is important)
  - What the reliability / availability needs to be
  - Conditioning on the characteristics of the functional challenge (support availability, phenomenology)
- In this case study, risk information has established the following:
  - Uncontrolled combustion of gases evolved in accidents can lead to containment failure and large radiological release
  - Potentially important sequences involve
    - Station blackout, affecting availability of power sources
    - Core melt phenomenology, affecting operability of systems in containment
    - Severe accident loads from phenomena other than combustion, influencing the impact of loads from combustion
- High-level statement of requirement: Prevent containment failure from uncontrolled combustion of gases in risk-significant scenarios.
- Begin application of guidelines by searching for monitorable parameters
  - Capability parameters (flowrates, heat removal rates, ...)
  - Reliability / Availability parameters

Establishing Capability Parameters



Guideline IA: Measurable (or calculable) parameters to monitor acceptable plant and licensee performance exist or can be developed

- Capability (of igniters)
  - Surface temperature
  - Distribution and number of units
    - Not related to ongoing performance; fixed property of design
  - Environmental qualification parameters
    - Not amenable to performance monitoring
- Reliability / Availability:
  - Functional reliability
  - Division reliability
  - Division availability
  - Unit reliability
  - Unit availability

- **Note: support systems need to be considered**

Guideline IB: Objective criteria to assess performance exist or can be developed

- Capability (of igniters)
  - Surface temperature, distribution and number of units
    - Parameters are established through model evaluations
  - Environmental qualification
    - Criteria can be developed from phenomenology
- Reliability / Availability:
  - Functional reliability is determined in light of functional challenge frequency and (e.g.) LERF guidelines
  - Given the functional reliability, and the design configuration, criteria can be established for division and unit level reliability & availability parameters

Guideline IC: Licensee flexibility in meeting the established performance criteria exists or can be developed

- Capability (of igniters)
  - Within a given technology, some limitations on flexibility would be implicit (Needed surface temperature determined by phenomenology, etc.)
  - Choice of technology could be allowed
- Reliability / Availability:
  - Flexibility exists in that there are different ways to achieve needed functional reliability
    - More redundancy in design means more (igniter) unit outages can be tolerated, different levels of unit reliability can be tolerated
    - Specifying availability averaged over a specified time period is in some ways more flexible than specifying an allowed outage time

Guideline ID: A framework exists or can be developed such that performance criteria, if not met, will not result in an immediate safety concern

- Capability parameters: For typical testing frequencies, degradation in monitored aspects of capability would be detectable within a short time
- Reliability / Availability Parameters: The reliability and availability needed in this function at most plants could be confirmed by monitoring (testing)
- The risk accepted when performance criteria are not met depends on
  - the length of time over which they are not met,
  - the likelihood of a functional challenge, and
  - the consequences of functional failure
- For this function, the combination of analysis, frequency of challenges to this function, and the LERF guidelines would be used to support acceptable time scales for detecting and addressing performance issues

Case Study 1
Summary

- Capability parameters
  - Aspects of capability such as environmental qualification are not amenable to performance-based treatment
    - Parameters and criteria exist, but it is not practical to confirm performance
  - Some capability parameters satisfy guidelines other than flexibility.
    - To achieve licensee flexibility, choice of technology needs to be allowed
- Reliability / Availability parameters satisfy all four viability guidelines
- This regulatory framework could be performance-based to a significant degree
- The guidelines were useful in evaluating the viability of a performance-based approach in this regulatory framework

Case Study 2

- this Case Study Is Fundamentally Different from the First Case Study - the Purpose Is Assessment Rather than Identification
- Performance-based Guidelines Are Applied to a Recently Revised Rule
- Assessment for this Case Study Is Limited to the Rule Level of the Regulatory Framework

Case Study 2

- Focus Application of the Guidelines on the Recent **Changes** Made to the Respiratory Protection Requirements (Subpart H of 10 Cfr 20)
- Viability Guidelines Were Thoroughly Applied to Three (3) Specific Changes to the Subpart H Requirements
- the Remaining Guidelines Were Applied to All the Changes to the Subpart H Requirements
- Do the Guidelines Support the Changes Made to the Requirements?

Case Study 2

Application of the Guidelines Is Only Made at the Rule Level

Rule Change	Rule Functionality	Guideline Application
Requirement to Include Non-radiological Safety Factors in Alara Analyses Increases Licensee Flexibility	Minimize Worker Risk Due to Airborne Hazards	Viable for Performance-based Approach Increase in Flexibility Makes the Revision More Amenable to a Performance-basing
Requirement to Meet Quantitative Fit Test Criteria and Testing Frequency Adds Prescriptive Requirements to the Rule	Ensure Proper Equipment Function	Limited Viability for Performance-based Approach Potential for an Immediate Safety Concern If Proper Fit Fails During Use Prescriptive Requirements Necessary to Ensure Accurate Dose Calculations
Revised Explicit Considerations for Respiratory Equipment Selection Neutral Impact on Licensee Burden	Ensure Selection of Proper Equipment	Limited Viability for Performance-based Approach Potential for an Immediate Safety Concern If Wrong Equipment Selected

Case Study 2

- the Remaining Guidelines Were Applied to the Changes to the Subpart h Requirements and Support the Changes Made to the Requirements

- Conclusion: the Results of Applying the Performance-based Guidelines Were Consistent with the Changes Made to the Subpart H Requirements

- this Case Study Demonstrated That Prescriptive Requirements Are Sometimes Necessary to Ensure the Accuracy of Performance Information

Interrelationships among Regulatory Initiatives

- Regulatory Initiatives Arise from Commission Direction, Operating Experience, Stakeholder Input, Staff Initiatives

- Screening Process Determines Whether to Pursue Initiative, and If So, with What Priority

- Elements of the Regulatory Framework Considered for Change as Part of the Initiative Are Identified

- Regulatory Approach Is Selected - (1) Risk-informed and Performance-based, (2) Risk-informed, (3) Performance-based, and (4) Traditional

    - this Selection Relies on Guidelines Developed as Part of this Performance-based Initiative and the Risk-informed Initiative

    - the Regulatory Approach May Differ from One Level of the Regulatory Framework to Another

    - Blend of Approaches Will Be Appropriate in Many Areas

- Procedures Applicable to Issuance of Regulatory Products (Backfit Analysis, Regulatory Analysis, Rulemaking Process) Remain Unchanged

Conclusions

- the Guidelines Direct the Analyst to Ask the Right Questions.
- Internal and External Stakeholder Support for the Guidelines Requires That Their Usefulness and Even-handedness Be Demonstrated in a Broad Range of Applications.
- Regulatory Improvement Using Performance-based Approaches Requires Consideration of the Entire Regulatory Framework.
- Selecting or Formulating Performance Parameters (Either Individually or in Groups) Is a Key Technical Challenge.
- Significant Progress Can Be Made Using a Simplified Expert Panel Process, Which May Be Followed up with More Rigorous and Documented Analysis.
- Improvements to Guidelines Will Be Considered as Experience Dictates.

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This report contains transcripts of papers presented at the Twenty-Eighth Water Reactor Safety Information Meeting held at the Bethesda Marriott Hotel, Bethesda, Maryland, October 23-25, 2000. The papers briefly describe the programs and results of nuclear safety research sponsored by the Office of Nuclear Regulatory Research, U.S. NRC. Papers concerning nuclear safety issues from U.S. government laboratories, the electric utilities, the nuclear industry, and from foreign governments and industry are also included.

The papers are printed in order of their presentation in each session and describe progress and results of programs in nuclear safety research conducted in this country and abroad. Foreign participation in the meeting included papers presented by researchers from France, Germany, Japan and Norway. The titles of the papers and the names of the authors have been updated and may differ from those that appeared in the final program of the meeting.

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