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K. C. Majumdar
H.P. Alesso
T.J. Altenbach

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ON FIRE RISK/METHODOLOGY FOR THE NEXT GENERATION OF REACTORS AND NUCLEAR FACILITIES*

K.C. Majumdar, H.P. Alesso, T.J. Altenbach
Lawrence Livermore National Laboratory
P.O. Box 808, L-196
Livermore, CA, USA 94550
(510) 422-5060

ABSTRACT

Methodologies for including fire in probabilistic risk assessments (PRAs) have been evolving during the last ten years. Many of these studies show that fire risk constitutes a significant percentage of external events, as well as the total core damage frequency. This paper summarizes the methodologies used in the fire risk analysis of the next generation of reactors and existing DOE nuclear facilities. Methodologies used in other industries, as well as existing nuclear power plants, are also discussed. Results of fire risk studies for various nuclear plants and facilities are shown and compared.

I. INTRODUCTION

A. Background

The possibility that fire could be a dominant risk contributor to nuclear power plants (NPP) and nuclear facilities (NF) was first recognized after the Browns Ferry cable tray fire in 1975.[1] Before that incident in which extensive failure of redundant and diverse safety-related systems occurred, fire protection system design criteria and methods were mainly guided by the standard practices in use for other industries. Since then, systematic deterministic fire design criteria evolved into the Appendix R to 10CFR50, and detailed design guidance was established in various industry standards such as IEEE-384 and NEPA.[2]

Methodologies for including fire in probabilistic risk assessments (PRAs) have been evolving, and a dozen NPPs have applied them during the last ten years. Many of these studies show that fire risk constitutes a significant percentage of external events as well as the total CDF (See Table 1).[3]

Recently the nuclear industry developed a structured and systematic methodology for the fire risk analysis using a screening probabilistic approach called *Fire Induced Vulnerability Evaluation (FIVE) Methodology*.[4-5] This methodology applies qualitative and quantitative screening techniques to estimate the frequency of dominant fire initiators and convolve these with the system model of the Level 1 PRA of the plant to come up with the CDF due to fire. The advantage of this screening method is that it does not depend on a highly uncertain, conventional fire PRA analysis. Instead, it uses deterministic design information, combining it with an existing probabilistic analytical system model of the plant and structured look-up tables. It utilizes the benefits of both qualitative and quantitative analyses.

FIVE is being applied to advanced reactors and the operating nuclear power plants. Nuclear non-reactor facilities may also employ a similar methodology, modified appropriately for the different missions of those facilities.

Since the Nuclear Regulatory Commission released the Reactor Safety Study, WASH-1400, in October, 1975, Probabilistic Risk Assessment (PRA) has been increasingly emphasized. Over 20 studies have been completed in the U. S. alone. The assessment of risk includes: identification of initiating events, measures of likelihood (including probability of core damage), and evaluation of consequences.[2-3]

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B. Objective

Our objective in this paper is to review and compare the important fire risk methodologies and the resulting contribution of fire to the overall risk of reactors and nuclear facilities.

II. METHODOLOGIES

Fire risk analysis in the non-nuclear industries uses probabilistic models to evaluate the risk-significant parameters. These models offer rational methods of dealing with randomness of fire risk and effectiveness of fire safety measures. Some of the methods include an exponential model of fire growth, probability distribution of damage, and a stochastic model of fire spread. The estimates provided by these models can be used with a currently available engineering method for probabilistic assessment of flame propagation. This approach is applied in textile, chemical, and other industries to estimate property damage.[6]

In the U. S., fire risk analysis methodology for nuclear power plants has been evolving and improved for over a decade. One of the recent methodologies,[4,7] combines engineering judgment, statistical evidence, fire phenomenology, and plant system analysis. The intended use of this analysis is for estimating the probability of a catastrophic effect on plant safety due to fire.

The latest probabilistic approach for fire risk analysis is the Fire Induced Vulnerability Evaluation (FIVE) Methodology by the Electric Power Research Institute (EPRI) for the nuclear power industry.[4]

This industry-developed, NRC-approved FIVE methodology was originally developed to evaluate the fire vulnerability of operating plants as required for NRC's *Individual Plant External Events Examination (IPEEE) Program.* This probabilistic screening methodology estimates fire-risk related to power operations. The main objective of this methodology is to evaluate the safe shutdown capability during and after fires in a nuclear plant, assessing whether the total risk due to fire remains below an acceptable level. Three main steps used to perform the evaluation are as follows:

1. Fire Area Screening (qualitative).
2. Critical Fire Compartment Screening (quantitative, using look-up tables developed from the analytical code, COMPBRNIIIe).
3. Plant Walk Down/Verification and Documentation.

Previous PRAs for nuclear plants used various analytical models for fire based on the WASH-1400 approach for calculation, but with different assumptions and upgraded designs as plants evolved. The evolution of fire models is discussed in References 2 and 5. The influence of various assumptions on the wide range of the estimates of CDFs due to fire are also discussed in those two documents. Finally, the COMPBRN code was developed that modeled a fire more realistically by the inclusion of detailed fire growth and propagation phenomena. Limited comparisons of the fire model in the latest version of COMPBRNIIIe have been validated with test results. FIVE also requires consideration of the additional following issues[8]:

1. Seismic/Fire Interactions.
2. Fire Barrier Qualifications.
3. Manual Fire Fighting Effectiveness.
4. Total Environmental Equipment Survival.
5. Control System Interactions.

In Step 1, fire areas are evaluated to screen out those areas that do not contain safe shut-down systems and components.

In Step 2, a plant level 1 PRA (i.e., System Analysis) is used for the quantitative screening. Then the frequency of fire initiating events are estimated and plant system models are developed for the fire scenario to estimate the total core damage frequency due to fire. Structured lookup tables are used in Step 2 for the quantitative evaluation.

In Step 3, a walkdown of the plant is made to verify the design conditions which have been used in Steps 1 and 2. A structured walkdown procedure and checklist are used for the walkdown. The walkdown reports are then documented for further design evaluation.

Non-reactor nuclear facilities of the Department of Energy have different missions and processes than reactors and therefore, require a different emphasis in the methodology. The traditional approach to fire related design and analysis at these facilities is by a deterministic analysis using the FIRAC code which is a transport model.[9] This model can calculate the airborne release of radioactive materials during the burning of contaminated combustible materials during a fire. FIRAC Code calculates both the transport of radioactive material and smoke particles. The radioactive material transport model uses radioactive source terms as input, that are calculated by the code FIRIN using phenomenological equations. Recently probabilistic risk assessment techniques are used for DOE non-reactor nuclear facilities for estimating the release of radioactive materials due to uncontrolled fire accidents. Dose exposures or the consequence analyses are also part of this assessment. The Replacement Tritium Facility (RTF) at the Savannah River site is an example of such a facility.

III. CORE DAMAGE FREQUENCY COMPARISON

In Figure 1, we present P(cd), the mean core damage frequency due to severe accidents. It shows the relative contribution of risk due to fire for the operating LWRs. In Table 1, comparison CDFs for dominant external events for operating plants are shown. In Table 2 CDFs for the dominant external events for advanced reactors are shown.

Comparisons of PRA values should be made with extreme caution since different models, assumptions, and degrees of sophistication in modeling were employed. In addition, we have only preliminary estimates available for a very few of the advanced reactor concepts. Nevertheless, they form the only available basis upon which any preliminary safety comparison can be attempted.

IV. DISCUSSION/CONCLUSION

An ALWR has completed a fire PRA using the FIVE Methodology. The ALWR (an ABWR) has three independent and physically separate safety divisions. Interaction between control room and remote shutdown panels in this plant has been minimized. ALWRs have much greater resistance to fire than earlier generations of LWRs due to design improvements. However, the Core Damage Frequency relative to the total CDF is still a significant percentage. This shows that fire protection design procedures are extremely important for the overall safety of the ALWRs. The key assumption in this analysis is that there is complete separation and therefore no interaction between systems and components of redundant safety divisions. In theory, it is possible to postulate such a separation, but in practice it is difficult to design and maintain such separation during a significant fire scenario. Verification of the design by walkdown, adherence to strict quality assurance measures and well-planned training for fire protection are the essential prerequisites for the validation of this key assumption.

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Table 1 External Events Core Damage Frequency (Event/R -Yr.) of operating LWRs

PLANT - LWRs	EARTHQUAKE	FIRE	REFERENCE/NOTE
Peach Bottom ·	5.5 X 10^{-5}(1)	2.0 X 10^{-5}(2)	(1) Average of LLNL and EPRI mean values from NUREG 1150, p8-6, Vol., June, 1989. (2) Mean value from *NUREG 1150*, p 8-6, Vol. 1, June, 1989.
Surrey -1	0.7 X 10^{-4}(1)	1.1 X 10^{-5}(2)	
Zion	5.6 X 10^{-6}	1.8 X 10^{-6} (3)(4)	(3) Fire Risk Analysis for Nuclear Power Plants: Methodological Developments and Applications, Risk Analysis, Vol.5, No. 1, 1985,-Kazarians, M., N. Siu, and G.Apostolakis.
Indian Point -2	4.8 X 10^{-5}	1.4 X 10^{-4}(3)(4)	
Indian Point -3	2.5 X 10^{-5}	9.6 X 10^{-5}(3)(4)	
Big Rock Point +	1.2 X 10-7(8)	2.3 X 1Γ -4(3)(4)	
Limerick	4.0 X 10^{-6}	2.3 X 10^{-5}(4)	(4) NUREG/CR-5042, UCID-21223, "Evaluation of External Hazard to Nuclear Power Plants in the United States, " C.Y. Kimura, R.J. Budnitz, Dec., 1987, p.3-4.
Seabrook	2.9 X 10^{-5}	2.5 X 10^{-5}(4)	
Oconee -3	6.3 X 10^{-5}	1.0 X 10^{-5}(4)	

Table 2 External Events Core Damage Frequency (Event/R-Yr.) of other reactors

PLANT	EARTHQUAKE	FIRE	REFERENCE/NOTE
ALWR CE Sys. .80+	150% of internal	--	Sys. 80+ about 7 X 10^{-7} Internal (estimated)
ALWR	--	~1.5 X 10^{-6}a)	6 X 10^{-6} Total CDF SECY-90-16 a) estimated
FBR CRBRP	3.2 X 10^{-5}(b)	5 X 10^{-7}	(b) Proceedings: International Topical Meeting on Probabilistic Safety Methods and Applications, Vol. 1: Session 1-8, EPRI NP-3912-SR, Vol., Feb. 1985, p. 62-7.
MHTGR	1.0 X 10^{-7}(c)	<1 X 10^{-7}(d) (est)	(c) Review of the standard MHTGR PRA, J.W. Minarcia, et al., ORNL, March 29, 1988, p. 25. (d) 1979 HTGR [2]
FACILITY			
Replacement Tritium Facility	TBD	TBD	

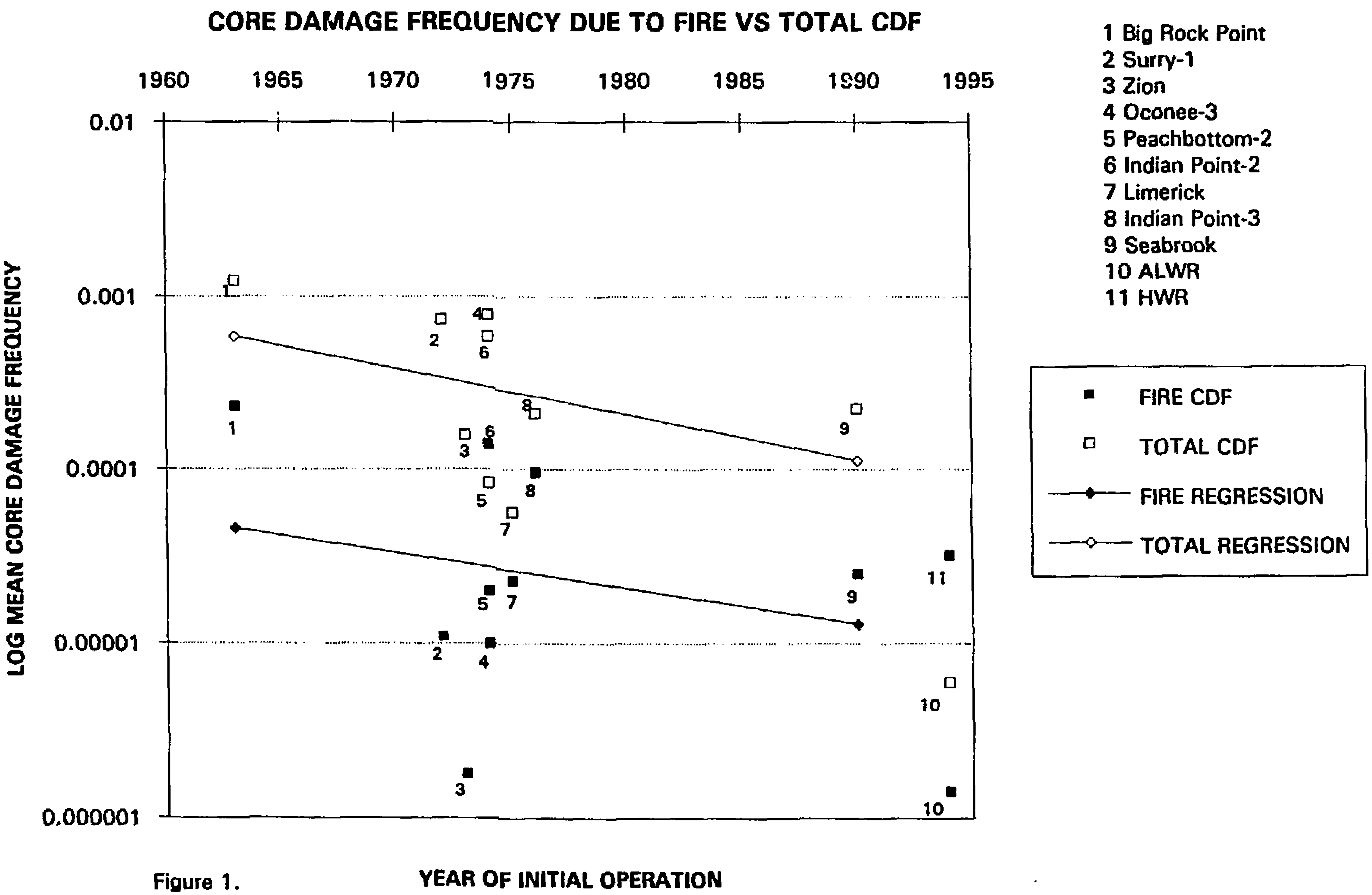
CORE DAMAGE FREQUENCY DUE TO FIRE VS TOTAL CDF
1960
1965
1970
1975
1980
1985
1990
1995
0.01
0.001
0.0001
0.00001
0.000001
LOG MEAN CORE DAMAGE FREQUENCY
YEAR OF INITIAL OPERATION
1 Big Rock Point
2 Surry-1
3 Zion
4 Oconee-3
5 Peachbottom-2
6 Indian Point-2
7 Limerick
8 Indian Point-3
9 Seabrook
10 ALWR
11 HWR
FIRE CDF
TOTAL CDF
FIRE REGRESSION
TOTAL REGRESSION

Figure 1.