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Analysis of Boron Dilution Transients in PWRs

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A study has been carried out with PARCS/RELAP5 to understand the consequences of hypothetical boron dilution events in pressurized water reactors. The scenarios of concern start with a small-break loss-of-coolant accident. If the event leads to boiling in the core and then the loss of natural circulation, a boron-free condensate can accumulate in the cold leg. The dilution event happens when natural circulation is re-established or a reactor coolant pump (RCP) is restarted in violation of operating procedures. This event is of particular concern in B&W reactors with a lowered-loop design and is a Generic Safety Issue for the U.S. Nuclear Regulatory Commission. The results of calculations with the reestablishment of natural circulation show that there is no unacceptable fuel damage. This is determined by calculating the maximum fuel pellet enthalpy, based on the three-dimensional model, and comparing it with the criterion for damage. The calculation is based on a model of a B&W reactor at beginning of the fuel cycle. If an RCP is restarted, unacceptable fuel damage may be possible in plants with sufficiently large volumes of boron-free condensate in the cold leg.

1. Introduction

1.1 Background

The U.S. Nuclear Regulatory Commission (NRC) has defined Generic Safety Issue 185 (GSI-185) “Control of Recriticality Following Small-Break LOCAs in PWRs.” [1] It refers to the situation in a pressurized water reactor (PWR) after a small-break loss-of-coolant accident (SBLOCA) when it is possible to have boiling in the core and condensation in the steam generators. There is no circulation under these conditions and the condensate, which is deborated, can fill the volume in the cold leg of the reactor coolant system (RCS) that is below the point where the cold leg empties into the reactor vessel. The concern is that either natural circulation would be re-established or a reactor coolant pump (RCP) would be restarted and the deborated slug of water would move into the core.

If the accident should occur early in the fuel cycle, there may be sufficient excess reactivity in the core for the deborated coolant to bring the core to criticality even though all the control rods have been inserted. The possible power excursion may be sufficient to cause severe damage to the core, even though the emergency core cooling system (ECCS) has successfully kept the core covered with coolant. It is this power excursion that is the basis for the generic issue. Unacceptable fuel damage is determined by the local fuel (pellet-average) enthalpy. The NRC currently uses 280 cal/g as the value which would lead to unacceptable fuel damage, although recent experimental research, in part supported by the NRC, has indicated that this failure limit may be significantly lower, particularly in high burnup fuel. It has been proposed that the acceptance criterion for fuel enthalpy might perhaps be reduced to as low as 100 cal/g [2].

This situation was expected to be most severe for a B&W designed plant with a lowered loop where the volume of the deborated slug of water could be quite large (~40 m^3) relative to other designs. As a result, the B&W Owners' Group (B&WOG) sponsored a study by Framatome Technologies [3, 4] to understand the event. This included looking into how the deborated slug might be created, how it might mix with highly borated water in the vessel, the consequences of the event in terms of core power, and the likelihood of a pump being restarted.

In order to resolve this generic issue the NRC developed a Task Action Plan (TAP). [5] A key component of the plan was to calculate the response of the core to the slug of deborated water that would be injected following either the restart of natural circulation or an RCP.

1.2 Objectives

The objective of this study was to provide coupled neutron kinetics and thermal-hydraulics analysis of several boron dilution events in order to assess the potential for fuel damage. The study considers the variables that are most important such as the volume of the deborated slug of water and its boron concentration and in this way considers the outcome in different plant designs.

2. Accident Analysis Methodology

2.1 Computer Codes Used

The PARCS (Purdue Advanced Reactor Core Simulator) code (Version v1.05) was used to simulate both the steady-state and transient reactor behavior of a B&W designed PWR. PARCS is a three-dimensional, two-group diffusion model using nodal methods [6]. It can be coupled with a thermal-hydraulics code to get a complete self-consistent simulation of the reactor core, or a simplified thermal-hydraulics model that is incorporated in the code can be used in a stand-alone mode.

To model the temporal and spatial dependence of the boron concentration and the thermal-hydraulic variables in the core, the RELAP5 code was used for this study. RELAP5 [7] is a generalized thermal-hydraulics code originally developed for modeling light water reactors, or similar systems. Although the governing equations are zero or one-dimensional, the three-dimensional thermal-hydraulic behavior in a reactor core can be modeled adequately by treating the core as an array of one-dimensional flow channels and control volumes that are connected to each other by junctions at the inlet and outlet plena.

2.2 Reactor Core Model

The Three Mile Island Unit 1 (TMI-1) B&W reactor at the beginning of a fuel cycle (BOC) was modeled with the above codes (Reference 8 gives details of the reactor). There are 177 fuel assemblies (see Figure 1) and 64 reflector assemblies modeled for the TMI-1 core. The PARCS model uses a 2x2 radial mesh in each fuel/reflector assembly, resulting in a total of 964 radial nodes. There are 28 axial neutronic nodes. The arrangement of control rod banks is shown in Figure 1. Banks 1, 2, 3, and 4 are safety banks that are inserted to shut down the reactor in the event of a reactor trip or a planned shutdown. Banks 5, 6, and 7 are regulating banks that are used in conjunction with the boron chemical shim to adjust the power level and maintain criticality over the fuel life cycle. Bank 8 contains axial power-shaping rods (APSRs). Rods 7A, 7B, and 7C in Figure 1 are individual control rods within Bank 7.

PARCS made use of a table look-up method for obtaining macroscopic cross section data for radially homogenized fuel assemblies. Data were available at five Doppler fuel temperatures, six moderator densities, and two boron concentrations within fuel assemblies with a specific burnup, and with the presence (or absence) of a control rod. The core of the TMI-1 PWR was modeled with 435 different compositions for unrodded fuel assemblies, and 195 compositions for rodded fuel assemblies [8]. There were additional compositions for the side, bottom and top reflector regions. The two-group cross section and assembly discontinuity factor data for each composition had been generated by the CASMO-3™ [9] lattice physics code at the BOC burn-up levels.

The shaded assemblies in Figure 1 are a symmetric octant of the core. To reduce the computational effort in the RELAP5 model, the symmetric fuel assemblies are lumped together in common thermal-hydraulic channels. The RELAP5 model represents the TMI-1 core as 30 parallel one-dimensional thermal-hydraulic channels joined by common mixing volumes at the inlet and exit. The bypass reflector assemblies are treated as a single channel. The RELAP5 thermal-hydraulic model uses a smaller number of axial nodes for the core than the PARCS thermal-hydraulics model (24 instead of 26). The axial reflectors and both the inlet and exit plena are represented explicitly.

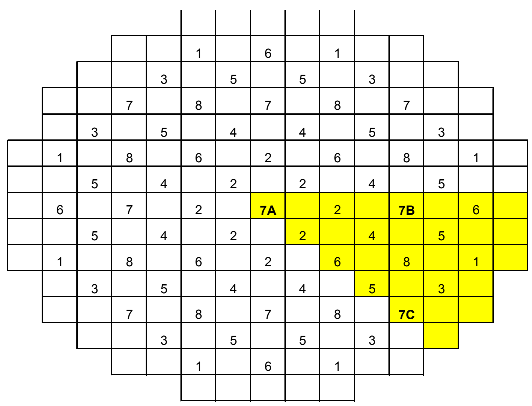

Fig. 1 Fuel Assembly Map With Control Rod Banks

2.3 Mixing Model

In order to provide appropriate boundary conditions to the PARCS/RELAP5 calculations, a model had been developed for mixing and transport of a deborated slug of water in the cold leg of a PWR [10]. Piping was modeled as plug flow volumes while the steam generator outlet plenum and the RCPs were modeled as back-mixed volumes. Although some mixing is expected in the piping, this approximation is reasonable for the limited length of piping being considered. The model has been validated against experimental data obtained from a test facility at the University of Maryland.

The model was applied to the present problem by using the geometry of steam generator tubes, steam generator outlet plenum, cold leg piping, and the RCP for several reactor designs. Two calculations were for the lowered loop B&W design corresponding to the model developed for the PARCS/RELAP5 calculations. In one case it was assumed that high pressure injection water (HPI) entering the vessel not only increased the water level in the core until natural circulation began but also caused the deborated slug to initially be pushed back up into the steam generator tubes. In the other case it was assumed that one of the two pumps in the cold leg being considered was restarted and mixing was more abrupt. Additional pump-on cases were done for the geometry of a three-loop Westinghouse (W) reactor (Beaver Valley) and a two-loop (four-pump) Combustion Engineering (C-E) reactor (Palisades).

To obtain the boron concentration at the vessel lower plenum, the first volume in the vessel modeled in PARCS/RELAP5, the assumption is made that there is no additional mixing in the vessel and hence, the concentration as a function of time is identical to that obtained by the ex-vessel model. This assumption is conservative. Once the slug of deborated water enters the lower plenum it mixes with the borated water and flows into the multiple channels that constitute the core inlet.

3. Calculational Results

3.1 Results for Natural Circulation

All results are shown starting at the time that the slug enters the lower plenum (time = 0). At this point the event may have been in progress for an hour or more so that by time-zero, the boron concentration in the core has increased to 2500 ppm due to ECCS injection and the core inlet temperature is assumed to have decreased to 227°C.

In Case 1 the boron concentration goes from 2500 ppm through the value 1165 ppm (the point at which the reactor is approximately critical) to zero in ~80 seconds representing a severe change in concentration but a modest rate of change. The reactor is initially shut down by ~\$15 at a power level of 10^{-7} of nominal power. The boron concentration and the resulting power (logarithmic scale) are shown in Figure 2. The first power spike, limited by fuel temperature feedback, has a pulse width of ~50 ms. The power peaks at almost five times nominal power and goes through a series of oscillations due to the competing effects of decreasing boron concentration and feedback from fuel temperature and moderator temperature and density changes. After 100 s the power is no longer significant, dropping below 1%.

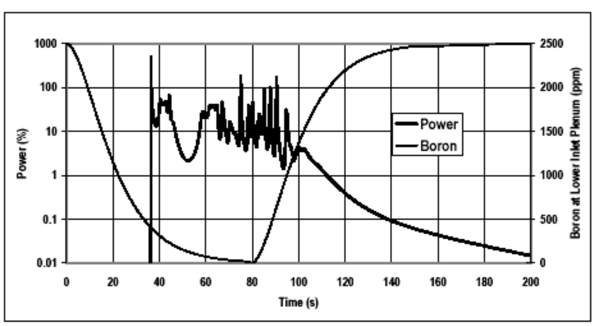

Fig. 2 Boron Concentration at Lower Plenum and Reactor Power (Case 1)

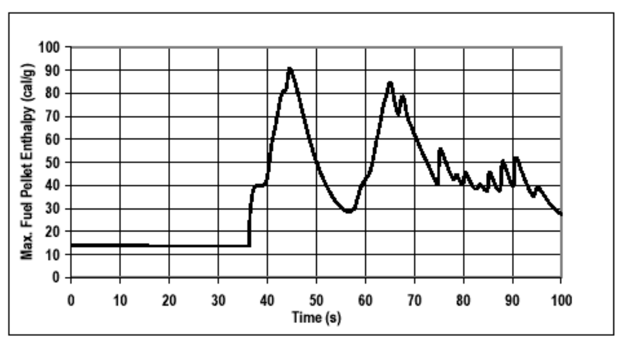

Fig. 3 Maximum Fuel Enthalpy (Case 1)

The quantity of most interest during this event is the maximum fuel enthalpy defined as the maximum over all nodes in the core, of the pellet radial average enthalpy. Since each node is approximately 15 cm long and averaged radially over an assembly, the true peak pellet-average enthalpy is actually larger by the peaking factor within that volume (~1.2). The peak fuel enthalpy as a function of time is given in Figure 3. The maximum over time is 90 cal/g or an increase in 73 cal/g relative to the initial fuel enthalpy. This corresponds to a maximum fuel centerline temperature of less than 2000°C. The initial rise in fuel enthalpy which takes place in the time frame of the initial power peak is only 25 cal/g. The peak values for fuel enthalpy occur in fresh fuel assemblies (adjacent to control rod position 7C in Figure 1) in the bottom 15 cm of the core.

The axial power distribution in the assembly with peak fuel enthalpy is shown at different times in Figure 4. The figure shows that the power distribution is peaked toward the bottom of the core as that is the region most affected by the boron dilution during the period when the power is greatest. Note too that the control rods do not cover the bottom 14.4 cm of fuel.

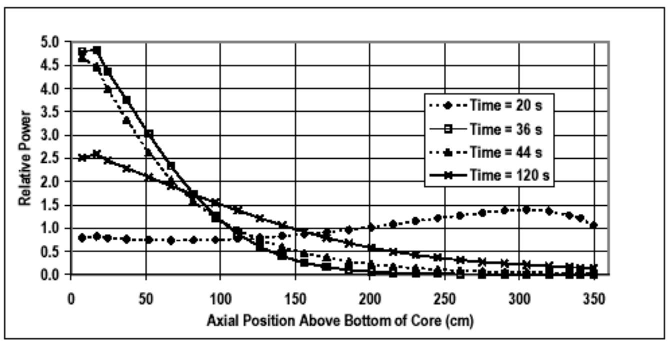

Fig. 4 Axial Power Distribution in Fuel Assembly 151 (Case 1)

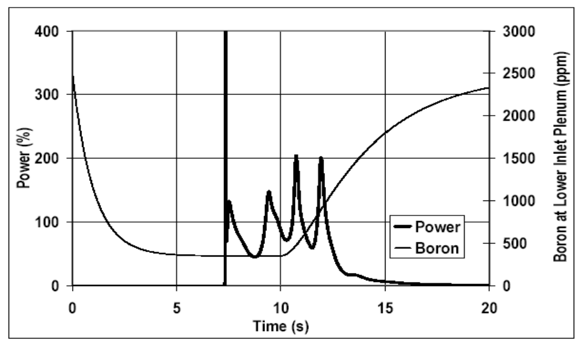

Fig. 5 Power and Boron Concentration (Case 2)

3.2 Results with Pump On

Case 2 differs from Case 1 in that one RCP has been restarted and hence, the flow rate at the time the slug enters the lower plenum is the result of natural circulation but then increases to 25% of full flow in 20 s. Although the rate of dilution is large, the potential minimum boron concentration, relative to the natural circulation Case 1, is less.

The core power as well as the boron concentration at the lower plenum is shown in Figure 5. Only 20 s of transient time are plotted as the event is over rapidly compared to the natural circulation transient (Case 1). The initial power spike of ~2700% is truncated on the plot. It is followed by four more power peaks ranging between 130% and 200%. By 15 s the power level drops below 10%, and continues to fall monotonically as borated water rapidly re-enters the core. Doppler and moderator feedback limit the power levels. The width of the first power pulse is ~17 ms.

The first power spike causes an initial jump in the temperature, and the fuel enthalpy rises by more than 30 cal/g. The subsequent power spikes continue the energy deposition, raising the fuel temperature to almost 2300°C (190 cal/g). The total increase in the fuel pellet enthalpy is more than

170 cal/g. Corresponding to this high fuel enthalpy is a peak fuel centerline temperature beyond the melting point of UO_2 (~2800°C). There is a period of nearly 5 s where the peak fuel centerline temperature exceeds the melting point in many fuel assemblies.

Two similar cases were run with different slug characteristics based on W and C-E designed plants. The geometry of the cold loop for these plants was used to determine the dilution rates. The calculations were still run with the TMI-1 model in PARCS/RELAP5 but the characteristics of the slug are meant to emulate what might occur in the other plants under the conservative assumption that there is sufficient condensate generated during the event to fill the cold leg. The response with the TMI-1 model is expected to be valid since the core characteristics, in terms of reactivity feedback and shutdown margin are expected to be similar enough in most PWRs to allow for a qualitative evaluation.

The results of pump-on calculations with these two slugs is quite different than for the B&W reactor. The deborated slug of water has a volume of only ~3.5 m^3 in these plants and the inlet plenum boron concentration is below 1165 ppm for only ~1 s as opposed to ~10 s in the B&W case. Hence, there is an increase in reactivity but not to the extent that the core goes critical before the effect of the slug passes and more borated water begins to enter the core.

4. Discussion of Results

4.1 Summary of Results

Table 1 provides information for each of the four cases including the maximum initial fuel enthalpy increase and the maximum fuel enthalpy throughout the transient simulation. Since the dynamic fuel behavior is different during the initial power surge which occurs in less than one second, than at later times after there is more time for conditions to equilibrate, it is necessary to look at both results.

For the case with natural circulation the initial fuel enthalpy increase (as a result of the initial power pulse), at the location where it is a maximum, is not significant (25 cal/g). The maximum fuel enthalpy during the event is 90 cal/g.

The situation with the restart of an RCP is more severe. At the time of the first power pulse more slug volume has entered the core with the pump on and the positive reactivity of the core is greater. With the pump on, the event is over in ~10 s whereas the slower transient lasts for more than 60 s. Furthermore the power level in the pump-on case remains at a higher level although it does go through the same type of irregular power peaking seen in the natural circulation case as the negative reactivity feedback mechanisms compete with the addition of reactivity due to dilution of the boron.

Fuel damage in Case 2 would certainly be indicated by the fact that the enthalpy is above the 170 cal/g limit currently imposed by the NRC in order to assess radiological consequences for boiling water reactors. At 190 cal/g the calculation also indicates centerline melting (temperature is calculated but melting is not part of the modeling) and fuel damage would be expected as a result. For high burnup fuel not only the fuel damage limit but potentially also the core coolability limit might be reached.

Table 1 Summary of Accident Analysis

Case	1	2	3	4
Mixing Model	DiMarzo B&W natural circulation	DiMarzo B&W pump-on	DiMarzo W pump-on	DiMarzo C-E pump-on
Slug Volume	42.5 m^3	42.5 m^3	3.6 m^3	3.5 m^3
Flow Rate	3%	25% (ramped)	33%	25%
Initial Fuel Enthalpy Rise cal/g	25	30	0	0
Maximum Fuel Enthalpy cal/g	90	190	17	17

The consequences of the event must be put into perspective by considering the probability of having a pump restart and how that probability changes the core damage frequency for the event. RCP restart or bump at B&W designed plants during a LOCA event is based on a set of criteria that assure that a potential slug of diluted water has had the opportunity to mix with borated water [11]. The frequency of occurrence of core damage with restart of natural circulation is 9×10^{-7}/reactor year according to the NRC [1] assuming a simple conditional probability for core damage given the restart, and 1.1×10^{-7}/reactor year according to Framatome assuming that conditional probability is unity. Without addressing the conditional probability for core damage, but just by assuming that the probability of restart of a pump is on the order of one or two orders of magnitude lower than the probability of restart of natural circulation, leads to a very low core damage frequency of $\sim 10^{-8}$/reactor-year.

For the cases with W and C-E designed plants, the fact that the cold leg volume that is below the entrance to the vessel is small makes these cases benign with respect to core damage. This is true with the assumption of restart of an RCP and hence, would also be the case with restart of natural circulation.

4.2 Uncertainty in Results

There are many sources of uncertainty in these calculations, some of which relate to the reactor model and some of which are specific to the boron dilution event being simulated. An analysis of the random errors in the reactor model has been assessed at RRC-KI based on their analysis with the BARS/RELAP5 code [12] for the event with restart of natural circulation. For nuclear parameters they did the analysis using integral parameters rather than starting from uncertainties in basic cross section data. Hence, they considered the effect on the fuel enthalpy of uncertainties in the moderator density reactivity coefficient, the worth of control rods, the boron reactivity coefficient, and the Doppler reactivity coefficient. Other sources of random error that they considered were the lower inlet plenum boron concentration, the fuel rod gap conductance, the lower inlet plenum flow rate, and additional smaller contributors. The result of their analysis indicated an uncertainty of ±90% in the maximum fuel enthalpy, corresponding to one standard deviation. The principal contribution to the aggregate uncertainty is the uncertainty in the moderator density reactivity coefficient which was assumed to be ±100%. This assessment does not account for uncertainties that might be present due to the codes being used to calculate power distributions with much larger gradients[1] than present when calculating reactivity coefficients.

The most significant source of bias is the boron concentration vs time assumed to flow into the lower plenum. The bias in the boron concentration comes from a) assuming the accident progresses long enough to fill the cold leg with boron-free condensate, and b) not including any mixing in the vessel or any mixing in the cold leg leading to the vessel except for that in the pump and at the steam generator outlet plenum such as due to the injection of HPI. The DiMarzo mixing model is conservative relative to a model developed by Framatome [3] and they consider their model to be very conservative. Amongst the several conservatisms that they claim to include in their analysis, one of the most important is that they do not model the vent valves present in these plants. "Had the vent valves been accurately modeled, they would have remained open during the restart of natural circulation, mixing core liquids with the coolant entering the downcomer from the cold legs. Because core liquid is highly borated, this provides a significant additional process by which the boron concentration of the deborate can be increased." [3] Indeed, they expect a realistic calculation would eliminate consideration of a prompt critical event for the natural circulation case.

Another source of error is the neglect of cross-flow in the core. If taken into account, this would reduce moderator temperatures in the hot channels thereby reducing the autocatalytic effect which causes hotter channels to have a reduced boron concentration because of their higher flow rate. However, the reduced moderator temperature also leads to reduced feedback in those channels, thereby compensating for the other effect and probably making the neglect of cross flow insignificant.

4.3 Conclusions

The boron dilution calculations described above were carried out to understand the consequences within the core under different assumptions. For the case with the restart of natural circulation it was shown that no fuel damage was expected. For the case with the restart of an RCP it was shown that fuel damage was possible under the conditions assumed for the B&W lowered loop design but not for the W or C-E designs. These consequences have to be put into the context of a very low frequency of occurrence.

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[1] Recall that there is a checkerboard pattern of control rods inserted and significant axial gradients (see Figure 4).

independent calculational results from Drs. A. Avvakumov and V. Malofeev at the Russian Research Centre - Kurchatov Institute, which helped assure the validity of the calculations done for this study. The guidance and support from Mr. H. Scott of the U.S. Nuclear Regulatory Commission is also greatly appreciated. Lastly, the authors thank Ms. S. Monteleone for assistance in the preparation of this paper.

References

1) Generic Safety Issue No. 185, "Control of Recriticality Following Small-Break LOCAs in PWRs," U.S. Nuclear Regulatory Commission, July 7, 2000.

2) R.O. Meyer et al., "A Regulatory Assessment of Test Data for Reactivity-Initiated Accidents," *Nucl. Safety, 37,* October-December 1996.

3) The B&W Owners Group Analysis Committee, Framatome Technologies, "Evaluation of Potential Boron Dilution Following Small Break Loss-of-Coolant Accident," Report 77-5002260-00 , Framatome Technologies, Lynchburg, VA, September, 1998.

4) The B&W Owners Group Analysis Committee, Framatome Technologies, "Evaluation of Potential Boron Dilution Following Small Break Loss-of-Coolant Accident," Final Report No. 47-5006624-00, Framatome Technologies, Lynchburg, VA, January 2000.

5) J.E. Rosenthal, Task Action Plan for Resolving Generic Safety Issue 185: "Control of Recriticality Following Small-Break LOCAs in PWRs," Memorandum to F. Eltawila, U.S. Nuclear Regulatory Commission, March 19, 2001.

6) H.G. Joo et al., "PARCS: A Multi-Dimensional Two-Group Reactor Kinetics Code Based on the Non-linear Analytic Nodal Method," PU/NE-98-26, Purdue University, School of Nuclear Engineering, September 1998.

7) INEL and EG&G, Idaho Falls, Idaho, "RELAP5/MOD3 Code Manual," NUREG/CR-5535, U.S. Nuclear Regulatory Commission, Washington, D.C., March, 1998.

8) K.N. Ivanov et al., "PWR Main Steam Line Break (MSLB) Benchmark; Volume I: Final Specifications," NEA/NSC/DOC(99)8, U.S. Nuclear Regulatory Commission and OECD Nuclear Energy Agency, April 1999.

9) N.K. Todorova and K.N. Ivanov, "Project Report on Task 1 of BNL Sub-contract Core Model - PARCS/RELAP5," Nuclear Engineering Program, Pennsylvania State University, August, 2000.

10) M. di Marzo, "Ex-vessel Transport and Mixing of a Deborated Slug in a PWR Primary Geometry," *Nuclear Engineering and Design, 210*, pp 169-175, 2001.

11) "Emergency Operating Procedures Technical Bases Document,"Framatome Technologies Technical Document No. 74-1152414-09, March 31, 2000.

12) A. Avvakumov, V. Malofeev, and V. Sidorov, "Analysis of a PWR Boron Dilution Accident Using the BARS-RELAP Code," NSI RRC KI 90-12/1-22-02, Russian Research Centre - Kurchatov Institute, January 2002. To be issued as a NUREG/IA by the U.S. Nuclear Regulatory Commission.