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# 13 ACCIDENT ANALYSES

## Introduction

This chapter presents analyses to show that the health and safety of the public and workers are protected in the event of an accident. This protection results from the facility design features, the Technical Specifications (Safety Limits, Limiting Safety System Settings, and Limiting Conditions for Operation), and the well-qualified and trained staff of NCNR. All of these combine to ensure that no credible accident could lead to unacceptable consequences to people or the environment.

The accident scenarios that need to be considered were defined for the original Safety Analysis Report (NBS, 1966, which is NBSR-9) for the NBSR and then redefined when the power level was increased to 20 MW (NBS, 1980). These scenarios take into account worst case assumptions expected to lead to the most severe consequences.

The present chapter differs from previous versions of the SAR primarily by the use of a new calculational methodology based on state-of-the-art computer tools for reactor physics and thermal-hydraulic analysis. The analysis (Carew, 2004, which is Appendix A to this SAR) was carried out by Brookhaven National Laboratory staff and staff from the NIST Center for Neutron Research.

The reactor physics studies of the NBSR core were performed with the three-dimensional Monte Carlo Neutron Photon (MCNP) code (Breimeister, 1997), and a geometric model initially developed at NIST. The final model included a plate-by-plate description of each fuel assembly, the unfueled mid-plane gap, beam tubes, cold neutron source, and tubular geometry of the shim safety arms, along with many other details of the reactor. For some of the studies, homogenization of partial regions was used for computational simplicity. Each of these cases was checked for the effect of the homogenization. The model was extensively benchmarked against measurements at NBSR of critical shim safety arm position, differential shim safety arm worth, regulating rod worth, performance of three different cold neutron source designs, neutron flux at beam tubes, and heat production in various structures. The MONTEBURNS code (Trellue, 1998), which links MCNP to ORIGEN, a code that calculates fission product production and decay, was used to calculate core inventory as a function of burnup. Models were created for beginning-, middle-, and end-of-cycle.

Time-dependent transient behavior of the reactor was calculated using RELAP5 (NUREG/CR-5535/Rev1) with a model that included the pumps, heat exchangers, fuel element geometry, and flow channels including both the six inner, and 24 outer, fuel elements. For some events MCNP results for power distributions were used to help obtain the critical heat flux ratio (CHFR, the ratio of actual heat flux to that required for film boiling). The Mirshak correlation was used to obtain the critical heat flux and a Monte Carlo method was used to obtain statistical results for the CHFR.

The BNL study (Appendix A) was used as the basis for many of the accident analyses that follow, and in many cases only the salient result is quoted below; in those cases, full details are contained in Appendix A.

# 13.1 Accident-Initiating Events and Scenarios

In this section all possible accident initiators are considered and the cases with the most limiting conditions are discussed. An analysis of these accidents and their consequences is provided in Section 13.2. The Maximum Hypothetical Accident (MHA) assumes fuel damage, and analyzes the consequences.

## 13.1.1 Maximum Hypothetical Accident (MHA)

The Maximum Hypothetical Accident is postulated as a complete blockage of flow to one element, leading to complete melting of the fuel plates. Such blockage is very unlikely, but is assumed for this analysis. The origin of the blockage is not identified; it is simply assumed. The consequences of this scenario bound the consequences of all partial blockages.

## 13.1.2 Insertion of Excess Reactivity

Detailed analysis shows that damage to the core from insertion of excess reactivity is not credible as a result of administrative controls, engineered safety features, and passive safety features in the NBSR. It should also be noted that addition of light water to the NBSR system provides negative reactivity in all concentrations (Section 4.5.2.2.3). The following reactivity insertion initiating scenarios are considered.

### 13.1.2.1 Step Reactivity Insertion

It is not credible that excess reactivity can be added to the NBSR by dropping a fuel element into an empty position in a critical core, since there are no empty positions. Further, refueling is only performed when the reactor is fully shut down with shim safety arms fully inserted. Further, only one element is ever moved at one time, so that an empty position could only arise from an element that had already been removed, making the reactor even further subcritical. When the core is being restored from the storage pool, it is possible to have empty locations in a nearly critical core, but procedural controls are in place to ensure that the shim safety arms are fully inserted when fuel is being moved. Having the shim safety arms inserted would preclude criticality even if the fuel were inserted improperly. No other mechanisms have been identified for a step (or very fast ramp) insertion of excess reactivity.

### 13.1.2.2 Ramp Reactivity Insertion

Two possible mechanisms for a ramp insertion of excess reactivity have been considered; the scenarios are given below. The result of an insertion of cold $D_2O$ was considered in NBSR-9, and shown to be a slow ramp insertion of less than 1% in 45 seconds. The consequences of such a scenario are clearly bounded by the two cases considered below.

#### 13.1.2.2.1 Startup Accident

For this initiating event, we assume that in violation of training and procedures, the reactor operator continues to withdraw the shim safety arms from the reactor at a rate equivalent to $5\times10^{-4}$ $\Delta\rho$ per second (a rate substantially in excess of the measured maximum rate at any shim safety arm position).

#### 13.1.2.2.2 Rapid Removal of Experiments

The total excess reactivity of all *removable* experiments in the NBSR is limited by the Technical Specifications to 1.3% $\Delta\rho$, with a limit on individual experiments of 0.5% $\Delta\rho$. Thus, the maximum credible excess reactivity insertion that could be caused by removal of a single experiment would be 0.5%$\Delta\rho$, and this could certainly not be accomplished in less than 0.5 s. In order to be compatible with earlier analyses in NBSR-9, a scenario is analyzed in which three experiments containing the maximum allowed reactivity (1.3%$\Delta\rho$) are removed in 0.5 s, which is a 2.6%$\Delta\rho$/s ramp.

## 13.1.3 Loss of Primary Coolant

A sudden loss of primary coolant from the NBSR is not credible. The main piping is located in protected areas, system pressures are low, and flow rates are small so that wear is not an issue. Nonetheless, the scenario assumes a major pipe break in the process room, which allows all of the primary coolant to drain from the reactor vessel into the process room located under the reactor while the reactor is operating at 20 MW.

## 13.1.4 Loss of Primary Coolant Flow

Five different scenarios for loss of primary coolant flow have been analyzed.

### 13.1.4.1 Loss of Off-Site Power

In this scenario, off-site power is lost, and the three primary coolant pumps trip. The reactor scrams on low flow.

### 13.1.4.2 Seizure of One Primary Coolant Pump

In this scenario, one of three primary pumps is assumed to seize up suddenly, imposing a rapid flow decrease, but the reactor scram as a result of low flow is delayed.

#### 13.1.4.3 Throttling of Primary Coolant Flow to Either the Inner or Outer Plenum

Because of the two-plenum configuration of the NBSR primary system (see Chapter 5), the possibility of inadvertent blockage of flow to one or the other plenum exists, presenting another scenario for a loss-of-flow transient.

#### 13.1.4.4 Loss of Both Shutdown Pumps

In this scenario, the loss of off-site power analyzed in Scenario 1 above is followed by a complete failure of all backup power sources (a highly unlikely event, as all systems undergo regular surveillance testing). The only core cooling after flow coast down is due to natural convection in the vessel and primary system.

## 13.1.5 Mishandling or Malfunction of Fuel

Four separate scenarios involving mishandling of fuel were extensively analyzed in NBSR-9, Addendum 1 (NBS, 1980), and shown to present no significant risks. These accidents were: a refueling accident involving a dropped element; dropping of a fuel element into the storage pool; dropping of a heavy object onto the fuel rack in the storage pool; and dropping of the spent fuel cask during a shipping operation. There has been no change in any of these accidents so the previous analysis remains valid. In addition to these scenarios, the possibility of an element being inserted into an incorrect position during refueling has now been analyzed, and shown to present no possibility of core damage. This analysis is presented in Section 13.2.5.

All fuel for the NBSR is subject to stringent quality control to ensure that there will be no “leaky” elements that could release fission products into the primary cooling system. In addition, if any element were to leak, the fission products would be detected immediately, and the faulty element would be identified and removed. This has only happened once in the operating history of the NBSR, and there were no releases to the atmosphere. The releases to the primary coolant were small, and the normal water treatment system quickly removed all traces of activity once the element was removed.

## 13.1.6 Experiment Malfunction

All experiments associated with the NBSR are carefully reviewed for hazards prior to being approved for construction and installation. Beam experiments external to the biological shield present a very small potential hazard to the reactor. Nevertheless, an experimental proposal must be prepared or amended before they can be installed or significantly modified. All proposals are reviewed in accordance with the Technical Specifications and Administrative Procedures. The Safety Evaluation Committee makes a recommendation to the Director of the NIST Center for Neutron Research, who has responsibility for final approval of any experiment. Thus, except for the reactivity issues addressed in Section 13.1.2, experiment malfunctions are not a credible threat to the core.

### 13.1.7 Loss of Normal Power

A Loss of Normal Power event is addressed in Section 13.1.4.1 above.

### 13.1.8 External Events

Damage to the core from external events, such as tornados, hurricanes, floods and earthquakes is not considered credible as a result of design features, administrative controls and the seismological and climatological characteristics of the site. Details are provided in Section 13.2.8.

## 13.2 Accident Analysis and Determination of Consequences

The NBSR is the only test reactor licensed and regulated by the Nuclear Regulatory Commission (NRC). As such, it is subject to the requirements of 10CFR100 in analyzing the consequences of any postulated accidents. It is used entirely as a source of neutrons for research in materials science, biology, chemistry, physics, and engineering. The power is limited to 20 MW, and there are no loop experiments or large-volume experiments (either permanent or removable) installed in the core. The reactor was designed with many passive safety features that limit the possibility of accidents resulting in fuel damage or radioactive releases (Section 1.2.3). The reactor is of the tank type, with a fully enclosed primary cooling system, moderator, and reflector. The reactor incorporates a passive emergency core cooling system. Reactivity decreases with increasing temperature. Reactivity also decreases with void formation in the primary coolant and with introduction of light water into the primary system. Thus, there is minimal potential for an accident with off-site radiological consequences. The exclusion zone is set at 400 m (entirely within the perimeter fence at the NIST site boundary) in Technical Specification 5.1, and all dose limits are calculated at this distance. This Emergency Planning Zone (EPZ) was chosen in accord with the guidance in NUREG-0849, Appendix II. The analyses for the present case were computed at this distance, and show that this is completely adequate for the EPZ.

### 13.2.1 Maximum Hypothetical Accident (MHA)

In this scenario, all primary coolant flow to one element is blocked by unspecified means. This would result in a rapid decrease in reactivity as the water in the element boiled and was expelled from the fuelled region of that element. As fuel temperature rises, local boiling of the moderator would occur, and cause power fluctuations. We assume that none of these leads to a shutdown, so that the element heats steadily until the fuel plates melt, releasing all of their fission products to the primary coolant. At this point, the reactor would be shut down for one of the following reasons:

- The fission product monitor would alarm shortly after the first release, leading to a manual reactor scram.
- As the fuel plates melt, fuel would drop out of the core region, leading to loss of reactivity and shutdown.

- The stack monitors in the effluent air exhaust would alarm, leading to an automatic major scram (TS Table 3.1).

Under any of these scenarios, normal ventilation is secured, confinement is isolated, and emergency ventilation would be automatically established by the high stack activity. This condition is assumed for the duration of the accident. At this stage, it is necessary to consider the timing and nature of the fission product release to the confinement building. Since the MHA does not involve a release of primary coolant, the important fission products are the noble gases and iodine (which may remain volatile at the temperatures that would be reached). The inventory of noble gas and iodine fission products in the most heavily irradiated element is given below in Table 13.1, as determined by the computer code ORIGEN2 (Croff, 1980).

All of the noble gas fission products would be released into the primary coolant and then, since they are insoluble in water, would quickly collect in the helium space at the top of the reactor vessel, which has a volume of approximately 0.7 $m^3$. The iodine releases require separate consideration. In past analyses, it was assumed, based on existing guidance, (DiNunno, 1962), (WASH-1400,1975), (Soffer, 1995) that 50% of the iodine would be released to the confinement building, with half of that available to the ventilation system. However, there has been extensive research into the iodine chemistry (Weber, 1992) that would take place in the aftermath of severe accidents. Although most of these analyses were aimed at power reactors, the results have also been used to develop an analysis (Weber, 1993) of a severe accident at the High Flux Isotope Reactor (HFIR) at Oak Ridge National Laboratory. The salient result of these studies for the present case is that for the temperatures that would occur in the NIST MHA, 99.9% of the iodine would be in the form of CsI, and would remain in solution in the primary coolant water. Radiolysis can transform CsI to $I_2$ gas, especially for low pH (up to 5% for pH = 5) situations, at high radiation doses. Nevertheless, consideration of these effects leads to the conclusion that less than 3% of the total iodine release will be present as $I_2$. Gaseous $I_2$ is soluble in water at up to 0.3 g/l at 298 K, with Henry's constant = 3.1 (dimensionless) (NIST, 2004). The large volume of primary coolant in the reactor vessel (which will remain below boiling temperature throughout the accident) will lead to very low $I_2$ concentrations, with correspondingly low vapor pressure, and $I_2$ will evaporate slowly into the helium space. This analysis makes the assumption that 3% of the initial iodine is released as $I_2$. Further, it makes the conservative assumption that the vapor pressure of $1.x10^{-9}$ bars, corresponding to the solution of this amount, is immediately available in the helium space at the top of the reactor vessel.

The preceding analysis describes the gaseous fission products that are immediately available in the helium space at the top of the reactor vessel. These will be released to the confinement building along with helium at a rate characteristic of the tightness of the primary system under emergency ventilation conditions (no normal building exhaust). This leak rate has been measured by observation of the increasing tritium levels in confinement during a prolonged shutdown of the building ventilation system for asbestos removal in February and March of 1989 (NIST, 1989). Table 13.2 shows the leak rates determined for the three areas of confinement. Exhaust rates to the stack from these spaces are then determined by the emergency ventilation system. The removal or release rates from each space are also shown in the Table 13.2.

These data provide the source term for estimating doses to the general public at the 400 meter exclusion radius, and to the staff in the building, under the conditions postulated for the MHA.

The doses to the public have been calculated following standard techniques. For doses resulting from the passage of radioactive clouds, the codes HOTSPOT (Lawrence Livermore National Laboratory, 2004) for short-term doses (first day), and CAP88(Environmental Protection Agency, 2004) for estimation of long-term (>1 day), have been used. The direct doses have been calculated following the methods used in NBSR-9, allowing for the depletion of the fission products in the building as the gases are released. The scattering from the air above the confinement building is calculated with SKYDOSE (Shultis, et. al., 1999). The iodine dose to the public is entirely negligible, as shown in Table 13.3. This is a direct result of the aqueous iodine chemistry, the mitigating effect of the filters, and the closed primary system. All dose components are small, and well within regulatory requirements.

To estimate the dose to the staff, the model of the release of noble gases and iodine developed above was used to calculate concentrations in rooms C-100 (the experimental floor) and C-200 (the operations level, where the control room is located) as a function of time spent in the area. The latest approved coefficients for immersion (Eckerman and Ryan, 1993) and inhalation were used to convert these concentrations into dose equivalents. The calculated dose to the staff is highest on C-200, where the reactor operators would be during an accident. The calculated dose on the first floor, where experimenters would be located, will be significantly lower as a result of the slower release rate to that area. To estimate doses, we assume immediate complete mixing (this is conservative, as the concentration will be highest in the middle of the room, while the control room is located nearer the outer perimeter walls). By procedure the operators would evacuate the building of all non-essential personnel immediately upon seeing the high readings of stack monitor and fission product monitor. They would then proceed to place the reactor in a safe condition, and leave themselves. For purposes of dose estimation, we assume that this takes 10 minutes, although it could be done more quickly. The doses are given in Table 13.4.

These calculated doses are based on conservative assumptions, and show that the reactor can be put into a safe condition and all personnel evacuated within the dose limits allowed for an emergency (exclusion from lifetime doses of 25 rem CEDE and 300 rem CDE to the thyroid). In practice, drills have shown that the occupants of C-100 can be evacuated from C-wing within 2-5 minutes without any difficulty, as has been demonstrated in prior drills. The reactor operators, who are stationed on C-200, would require more time to ensure that systems are properly secured, but would be able to evacuate within 10 minutes, leaving the reactor in a completely safe configuration. If required, operators could re-enter the confinement building to perform surveillance or other tasks. In fact, they could remain in C-200 for up to 25 minutes without exceeding the 25 rem CEDE emergency dose limit.

The above calculation of the estimated dose to the staff was re-examined using a more realistic yet still conservative assumption for the flow blockage of a fuel element. A screen located upstream of the core has a 0.25 inch (0.635 cm) square mesh, which could allow a thin piece of material 0.35 inch (0.89 cm) wide to pass and enter a fuel element. Assuming, conservatively, that this piece is 3 inches (7.62 cm) long, and also, conservatively, that it positioned itself so that it completely blocked flow to two channels on either side of a single plate, then only one plate (out of 34) will fail. In this case, the doses calculated above will be reduced by a factor of 34, or to 3% of the values shown in Table 13-3.

On the basis of these calculations, the projected doses from the MHA are acceptable both for the general public and for the staff.

## 13.2.2 Insertion of Excess Reactivity

### 13.2.2.1 Step Reactivity Insertion

This scenario has not been analyzed since there is no credible initiating scenario.

### 13.2.2.2 Ramp Reactivity Insertion

The accidents associated with the two scenarios described in 13.1.2 have been analyzed using a point kinetics model. The reactivity worth of the shim safety arms is shown as a function of angle in Figure 13.1. In scenarios involving excess reactivity insertion, the initial rate of insertion of negative reactivity following a scram is a critical parameter. Figure 13.1 shows clearly that this rate is lowest (for the operating range of the shim safety arms) at end-of-cycle (EOC), when the shim safety arms are fully withdrawn. Thus, the EOC case is limiting for the maximum reactivity insertion, but the SU and EOC cores had about the same behavior for the startup accident.

13.2.2.2.1 Startup Accident

This accident has been analyzed, as described in Appendix A. Calculations were performed for both the startup core and the EOC core, and the results at EOC produced the larger energy excursion. For these calculations, the reactivity insertion rate was assumed to be $5\times10^{-4}$ Δk/s, a rate substantially in excess of the measured maximum rate at any shim safety arm position, and particularly conservative at EOC. The scram was assumed to occur at 130% of full power (the TS LSSS).

The shim safety arm insertion was assumed to be described by:

$$\Delta\theta = a(t-\delta)^2$$

where

$a=248.9\ ^{\circ}/s^2$,

$\delta = 0.0983$ s, and

t = time after scram initiation.

This implies a time of 0.241 s to insert the shim safety arms 5º, substantially in excess of the 0.220 s that is the limit specified in the TS surveillance tests. Conservatively no temperature or other reactivity feedback mechanism was included in the calculation. Using these assumptions, RELAP5 was used to study the transient behavior, with the result shown in Figure 13.2. This scenario results in a Minimum Critical Heat Flux Ratio (MCHFR) greater than 1.7, providing ample margin to ensure that no fuel damage will result. This result is conservative, for reasons that are discussed in Chapter 4 and in section 13.2.2.2.2 below. The startup accident in the SU core, however, had a lower MCHFR, 1.55, than the EOC case, even though the peak power was somewhat lower, as seen in Figure 13.3. The SU hot spot has a higher heat flux whereas the power excursions were very similar.

13.2.2.2.2 Rapid Removal of Experiments

This scenario is not credible since it assumes that three separate removable experiments can be extracted from the core simultaneously in 0.5 s, and that this results in the insertion of the maximum 1.3% $\Delta\rho$ excess reactivity. In order to accomplish this insertion, a very careful plan and procedure for simultaneous action by three operators would be required. Nevertheless, this postulated accident has also been analyzed with the following assumptions:

- Initial power = 20.4 MW
- Reactor power scram occurs at the LSSS of 26 MW (130%)
- Negative feedback from increasing fuel and coolant temperatures is neglected
- Shim safety arm motion as in Section 13.2.2.2.1
- Prompt neutron lifetime of 650 μs

Two cases were analyzed, one for the startup core and the second for the equilibrium EOC core, and both are described in detail in Appendix A. The limiting case occurs at EOC, when the shim safety arms are fully withdrawn. The results for power and MCHFR as functions of time are shown in Figure 13.4. The lowest value of the MCHFR observed occurs in the outer plenum, and is greater than 1.18, providing a substantial margin against fuel damage.

This estimate is conservative, for the following reasons:

- The heat fluxes are estimated on the basis of the fission density, which assumes that all of the energy is deposited locally. However, 14 % of the fission energy will be deposited uniformly throughout the core, in other structures or in the moderator.
- The original model was used for heat fluxes, which is shown to be conservative in Chapter 4.
- The effect of three dimensional heat transfer from the hot stripe and hot spot to neighboring unfueled regions was neglected. Calculations have been done using finite element heat transfer to show that this effect reduces heat fluxes at the hot spot by as much as 10%. The effects of cooler water in the channel next to the hot channel are ignored.
- The 650 μs prompt neutron lifetime used is conservative; the MCNP calculations presented in Chapter 4 indicate a value closer to 800 μs.
- The scram was assumed to occur at 130% of power, rather than the actual setting of 125% of power.

Therefore, this transient, which is in itself unrealistically conservative, will not lead to any fuel damage or to the release of fission products.

### 13.2.3 Loss of Primary Coolant

This scenario is extremely unlikely, for the reasons given in Section 13.1.3. However, for purposes of analysis, we assume a major pipe rupture that drains the entire contents of the reactor vessel, approximately 3,000 gal (11 $m^3$), into the process room. The primary coolant is trapped there by a dam built for the purpose, resulting in a pool with a surface area of approximately 1080 $ft^2$ (100 $m^2$). The reactor scrams immediately on a loss–of-flow signal. The operation of the NBSR emergency core cooling system, for which initial action is totally passive, is fully described in Chapter 6. Primary coolant, contained above the core in the Inner Reserve Tank (IRT), drains to a distribution pan that directs the coolant to individual elements for several minutes when needed. No action is required to initiate this flow. Operation of a single valve adds the capacity of the 3,000 gallon (11 $m^3$) $D_2O$ Emergency Cooling Tank located on the operations floor 30 ft (9.1 m) above the core. Thus, with only one operator action (which can be accomplished at any time in the first 20 minutes), the core is fully protected for several hours. During this time, a system already in place can be started, and lost primary water would be pumped from the dammed area in the process room up to the $D_2O$ Emergency Cooling Tank, providing virtually unlimited cooling time. Alternatively, if needed, through the addition of a single spool piece, light water can be piped into the system to provide cooling. The operation of the emergency cooling system has been analyzed in Appendix A, where it has been shown that the water will flow into the elements from the top for over 20 minutes. With the cooling provided by this system, the temperature of the clad will remain well below any blistering temperature. Thus, no fission products will be released during this accident. However, the primary water will contain tritium as a result of neutron capture in the heavy water, and the radiological consequences of this needs to be computed.

For analysis purposes, the following conservative assumptions are made:

- The tritium concentration in the primary coolant is at the maximum level permitted by the Technical Specifications (5,000 μCi/ml).
- After the break, emergency ventilation is immediately established.
- The process room is not isolated from the emergency ventilation system (ACV-10 is left open).
- The Emergency Ventilation System pulls the maximum design flow of 15 cfm ($7.1 \times 10^{-3}$ $m^3/s$) from this area.
- Equilibrium between the spilled heavy water at an assumed temperature of 108˚F (42˚C) and the air in the process room is established immediately.

With these assumptions, we calculate the rate of tritium release to the stack:

$$R = F\rho_{D2O}C$$

where:
$F$ = Flow rate = $7.08 \times 10^{-3}$ m$^3$/s,
$\rho_{D2O}$ = mass of $D_2O$ per m$^3$ at saturated vapor pressure = 55 g/m$^3$, and
$C$ = Tritium Concentration = 5,000 μCi/ml = $4.5 \times 10^{-3}$ Ci/g.

Or,

$$R = 1.8 \times 10^{-3} \text{ Ci/s}.$$

Using this release rate, the effluent concentrations have been calculated for a variety of weather conditions, using three different EPA codes (COMPLY, SCREEN3, and CAP-88). These codes have different levels of conservatism built into them, roughly in the order that they are listed with COMPLY being the most conservative. For all of the codes listed, and weather conditions used, the effluent concentration at or beyond the 400 m boundary is less than 1000 nCi/m$^3$. This last value was found for extremely stable conditions and low wind speeds, which could not persist over any significant length of time. It should be noted that any release would be terminated within 24 hours, as remedial measures (pumping water into tanks, closing ACV-10, covering spilled water with plastic) would be taken immediately. Taking these time factors into account, no individual would receive as much as 0.2 mrem total dose even if they stood at the boundary throughout the release. If the entire inventory were to leak out in this manner, a person at the site boundary would receive less than 6.5 mrem (calculated using COMPLY), or 6.5% of the permissible annual dose to the general public. This last calculation assumes average weather conditions over the year, as measured at Ronald Reagan Washington National Airport, and assumes the entire inventory in the vessel is released. Since this accident would not result in exposures approaching 10CFR20 limits, there are no serious off-site consequences.

The primary coolant is confined to the process room where the tritium levels are determined by the vapor pressure. For the conditions analyzed, this will result in a concentration approaching $1.25 \times 10^{-4}$ DAC. Access to this area is always strictly controlled. If prolonged access were required, special provisions would be implemented to control exposure to acceptable levels.

## 13.2.4 Loss of Primary Coolant Flow

Four scenarios have been given for an accident of this type, and all were analyzed (see Appendix A). None of these scenarios led to fuel damage, and the minimum value of the CHFR during the transients analyzed was found to be 2.19 for the case of loss of off-site power (Section 13.1.4.1).

The results for the fourth scenario are of particular interest, since they cover the case of loss of forced flow and onset of natural convection cooling (it should be noted that primary coolant flow is upward through the elements in the NBSR, so that no flow reversal is required to establish convection cooling). The results of the transient analysis of coolant conditions are shown in

Table 13.5 below. It should be noted that the beginning of this transient is exactly the same as that for the loss of off-site power (see Appendix A). There is some oscillation of flow as the natural convection loop begins to cool the reactor (between 40-160 seconds), but the situation rapidly stabilizes, and there is no possibility of fuel damage.

### 13.2.5 Mishandling or Malfunction of Fuel

During refueling of the NBSR, all 30 elements are moved; four are removed to the storage pool, the remaining 26 are moved to new positions, and four new elements are added as part of a carefully planned and executed fuel management program. During this operation, there are always two operators at the reactor top, one to move the element, and the other to verify that the move is correct. In addition, there is an operator in the control room, who also verifies and records each move. These procedures make it very unlikely that an element could be loaded into an incorrect location. Nevertheless, the following is an analysis of the case in which, in spite of all procedures, a fresh element is located in a higher flux location than planned, leading to a higher heat load.

The possible power peaking was calculated with MCNP by sequentially switching one fresh element with one of the 26 partially burned elements in the startup core. The result of this calculation allowed selection of the worst possible case for further analysis. Details of the calculation are given in Appendix A, and summarized in Table 13.6. The minimum value of the CHFR is 2.0, and therefore no fuel damage is anticipated (see Appendix A).

### 13.2.6 Experiment Malfunction

The only scenario of concern here is for an experiment internal to the reactor biological shield. The most significant malfunction of an experiment of this type has already been analyzed as a ramp insertion of excess reactivity. This accident bounds all reactivity effects of experiments, including flooding of beam tubes. All experiments involving explosive or corrosive materials must be reviewed in detail by the Safety Evaluation Committee, and approved by the Director, NCNR, before implementation. Quantities of explosives to be irradiated in the core are strictly limited to amounts for which any explosion can be totally contained within the experiment packaging. Therefore, damage to the core from an experiment malfunction is not credible.

### 13.2.7 Loss of Normal Power

This accident is addressed in Section 13.2.4 above.

### 13.2.8 External Event

The NBSR is located in a zone of low seismic activity. The building and reactor systems have been analyzed and shown to be able to withstand the stresses generated by a 0.1 g earthquake loading (NBS, 1966b). The probability of an earthquake resulting in accelerations larger than 0.08 g is less than 2% in 50 years (Figure 13.5).

The confinement building was designed to withstand the forces generated by winds of up to 100 mph, substantially faster than the largest wind ever recorded at Ronald Reagan Washington National Airport (76 mph during passage of Hurricane Hazel, October 1954).

The computed recurrence interval for a tornado at the NIST site is approximately 2000 years. The NBSR is immediately shut down if NIST Security notifies the Control Room that a tornado or other major weather hazard is approaching the site. This action is specified in the Emergency Instructions Manual. Further, if a tornado is sighted on the NIST site, a Notification of Unusual Event is declared.

During unsettled weather conditions, Control Room personnel monitor all weather alerts.

Therefore, none of these scenarios pose a significant threat to the reactor. Further, it is difficult to envision any accident resulting from such a scenario that would have consequences exceeding those already analyzed.

## 13.3 Summary and Conclusions

This chapter has presented the results of a number of conservative analyses of potential accidents related to operation of the NBSR. No credible accident results in core damage. Nonetheless, the MHA analysis assumes core damage. Even in this case, the resultant consequences are well within the limits of 10CFR100, which applies to Test Reactors (and below 10CFR20 limits for the general public). Therefore, operation of the NBSR will present no undue hazard to any member of the general public or to the staff.

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**Table 13.1: Maximum Iodine and Noble Gas Fission Product Inventory in Fuel Element After Eight Cycles**

| Isotope | Inventory (Ci) | Half Life (s) | | Isotope | Inventory (Ci) | Half Life (s) |
|---|---|---|---|---|---|---|
| $^{130}I$ | 9.36E+02 | 4.45E+04 | | $^{83m}Kr$ | 2.94E+03 | 6.59E+03 |
| $^{130m}I$ | 3.29E+02 | 5.40E+02 | | $^{85}Kr$ | 7.91E+01 | 3.41E+08 |
| $^{131}I$ | 1.59E+04 | 2.89E+04 | | $^{85m}Kr$ | 7.01E+03 | 1.58E+04 |
| $^{132}I$ | 2.41E+04 | 8.26E+03 | | $^{87}Kr$ | 1.42E+04 | 4.56E+03 |
| $^{133}I$ | 3.72E+04 | 7.49E+04 | | $^{88}Kr$ | 2.00E+04 | 1.01E+04 |
| $^{133m}I$ | 7.11E+02 | 9.00E+00 | | $^{131m}Xe$ | 1.78E+02 | 1.03E+06 |
| $^{134}I$ | 4.20E+04 | 3.15E+03 | | $^{133}Xe$ | 3.66E+04 | 3.88E+04 |
| $^{134m}I$ | 2.41E+03 | 2.16E+02 | | $^{133m}Xe$ | 1.10E+03 | 1.89E+05 |
| $^{135}I$ | 3.47E+04 | 2.37E+04 | | $^{135}Xe$ | 9.42E+02 | 3.32E+04 |
| | | | | $^{135m}Xe$ | 6.26E+03 | 9.17E+02 |
| | | | | $^{137}Xe$ | 3.31E+04 | 2.29E+02 |
| | | | | $^{138}Xe$ | 3.44E+04 | 8.45E+02 |

**Table 13.2: Leak Rates To Confinement And Release Rate To Stack**

| Confinement Area | Volume ($m^3$) | Leak Rate ($m^3$/s) | Removal Rates ($m^3$/s) | Removal Rates (cfm) |
|---|---|---|---|---|
| First Floor (C-100) | $4.5 \times 10^{3}$ | $1.2 \times 10^{-6}$ | $9.4 \times 10^{-3}$ | 20 |
| Second Floor (C-200) | $8.3 \times 10^{3}$ | $8.1 \times 10^{-6}$ | $18. \times 10^{-3}$ | 40 |
| Process Room | $2.0 \times 10^{3}$ | $2.3 \times 10^{-6}$ | $7.1 \times 10^{-3}$ | 15 |

**Table 13.3: Dose to an Individual at the Edge of the 400 meter Exclusion Zone After the Maximum Hypothetical Accident**

| Doses (in mrem) to a person standing at the edge of the exclusion zone (400m) | | 0-2 Hours | 2-24 Hours | 1-30 Days | Total |
|---|---|---|---|---|---|
| Radiation from fission product decay | Direct | 0.0 | 0.0 | 0.1 | 0.1 |
| | Sky Shine | 0.0 | 0.0 | 0.2 | 0.2 |
| | Total | 0.0 | 0.0 | 0.3 | 0.3 |
| Iodine Releases | CEDE[1] | 0.0 | 0.0 | 0.0 | 0.0 |
| | Thyroid | 0.0 | 0.1 | 0.0 | 0.1 |
| Noble Gas Cloud (max) | Slightly Stable Conditions[2] | 1.0 | 3.4 | 2.0 | 6.4 |

[1] The dose given is the maximum Committed Effective Dose Equivalent (CEDE) at a point on or outside the 400 m boundary.
[2] The wind speed is assumed constant at 1 m/s, with allowance for meandering (averaged over 120 min) but no wind shifts, Pasquill diffusion category E. The dose due to the cloud outside the boundary is then a maximum for this stability condition. However, such conditions could not persist for even 12 hours, so these numbers are large over-estimates. Actual doses would be 2-100 times lower.

**Table 13.4: Calculated Dose to Staff as a Result of the MHA**

| Location | Type of Dose | 10 Minute Dose (Rem) |
|---|---|---|
| C-100 | Whole Body (TEDE) | 1.07 |
| | Thyroid (CDE) | 0.01 |
| C-200 | Whole Body (TEDE) | 4.06 |
| | Thyroid (CDE) | 0.02 |

**Table 13.5: Transient Conditions of Primary Coolant Following Loss of All Primary Pumps (Main and Shutdown)**

| Times<br>s | Reactor Power<br>MW | Primary Flow Rate<br>gpm (x15.85 l/s) | Inner Plenum Flow<br>gpm (x 15.85 l/s) | Outer Plenum Flow<br>gpm (x 15.85 l/s) | Coolant Temperature °K | | |
|---|---|---|---|---|---|---|---|
| | | | | | Inner Plenum | Outer Plenum | Reactor Outlet |
| 0.0 | 20.40 | 8700.0 | 2300.0 | 6400.0 | 316.6 | 316.6 | 324.7 |
| 20.0 | 1.06 | 134.6 | 2.1 | 130.4 | 316.6 | 316.6 | 324.7 |
| 40.0 | 0.87 | -9.1 | -6.4 | -3.4 | 329.5 | 316.6 | 324.7 |
| 60.0 | 0.77 | 2.0 | -22.4 | 24.2 | 348.0 | 328.1 | 324.7 |
| 80.0 | 0.71 | 7.7 | 4.5 | 3.1 | 338.5 | 330.3 | 324.7 |
| 100.0 | 0.66 | 43.9 | 30.6 | 13.1 | 336.6 | 335.8 | 324.7 |
| 120.0 | 0.63 | 61.5 | 10.1 | 51.2 | 323.4 | 333.0 | 324.7 |
| 140.0 | 0.61 | 67.5 | 11.1 | 56.1 | 321.3 | 323.7 | 324.7 |
| 160.0 | 0.59 | 59.4 | 12.2 | 46.9 | 319.3 | 319.9 | 324.7 |
| 180.0 | 0.57 | 44.0 | 8.3 | 35.5 | 318.7 | 318.4 | 324.7 |
| 200.0 | 0.56 | 38.7 | 8.5 | 30.0 | 318.3 | 317.7 | 324.7 |
| 220.0 | 0.55 | 41.5 | 8.1 | 33.2 | 318.1 | 317.3 | 324.6 |
| 240.0 | 0.54 | 41.9 | 8.7 | 33.0 | 317.9 | 317.0 | 324.6 |
| 260.0 | 0.53 | 37.6 | 7.7 | 29.8 | 317.7 | 316.8 | 324.6 |
| 280.0 | 0.52 | 35.1 | 7.3 | 27.6 | 317.6 | 316.7 | 324.6 |
| 300.0 | 0.51 | 35.8 | 7.3 | 28.4 | 317.5 | 316.7 | 324.6 |
| 320.0 | 0.50 | 36.3 | 7.4 | 28.8 | 317.4 | 316.6 | 324.6 |
| 340.0 | 0.50 | 34.7 | 7.1 | 27.5 | 317.3 | 316.6 | 324.6 |
| 360.0 | 0.49 | 33.0 | 6.8 | 26.1 | 317.2 | 316.6 | 324.6 |
| 380.0 | 0.49 | 32.7 | 6.7 | 25.9 | 317.2 | 316.5 | 324.6 |
| 400.0 | 0.48 | 32.9 | 6.7 | 26.1 | 317.1 | 316.5 | 324.6 |
| 420.0 | 0.48 | 32.3 | 6.5 | 25.6 | 317.1 | 316.5 | 324.6 |
| 440.0 | 0.47 | 31.2 | 6.3 | 24.8 | 317.0 | 316.5 | 324.6 |
| 460.0 | 0.47 | 30.7 | 6.2 | 24.3 | 317.0 | 316.5 | 324.6 |
| 480.0 | 0.46 | 30.5 | 6.2 | 24.2 | 316.9 | 316.5 | 324.6 |
| 500.0 | 0.46 | 30.2 | 6.1 | 24.0 | 316.9 | 316.5 | 324.6 |

**Table 13.6: Minimum Critical Heat Flux Ratio for Misloaded Fresh Fuel**

| | Inner Core | | Outer Core | |
|---|---|---|---|---|
| | **EQ-BOC**[1] | **Fresh Fuel**[2] | **EQ-BOC** | **Fresh Fuel** |
| **Relative Radial Power** | 1.07 | 1.68 | 1.16 | 1.51 |
| **Peak Heat Flux ($W/m^2$)** | $1.4x10^6$ | $2.2x10^6$ | $1.7x10^6$ | $2.2x10^6$ |
| **Coolant Temperature**[3] **(°K)** | 323.1 | 326.9 | 328.0 | 331.5 |
| **Critical Heat Flux ($W/m^2$)** | $5.6x10^6$ | $5.5x10^6$ | $4.6x10^6$ | $4.5x10^6$ |
| **Minimum Critical Heat Flux Ratio** | 4.0 | 2.5 | 2.7 | 2.0 |

Notes:
[1] EQ-BOC refers to the equilibrium core beginning of cycle conditions.
[2] Fresh Fuel refers to the worst case of misloaded fuel.
[3]Core inlet temperature is 316.6 K.

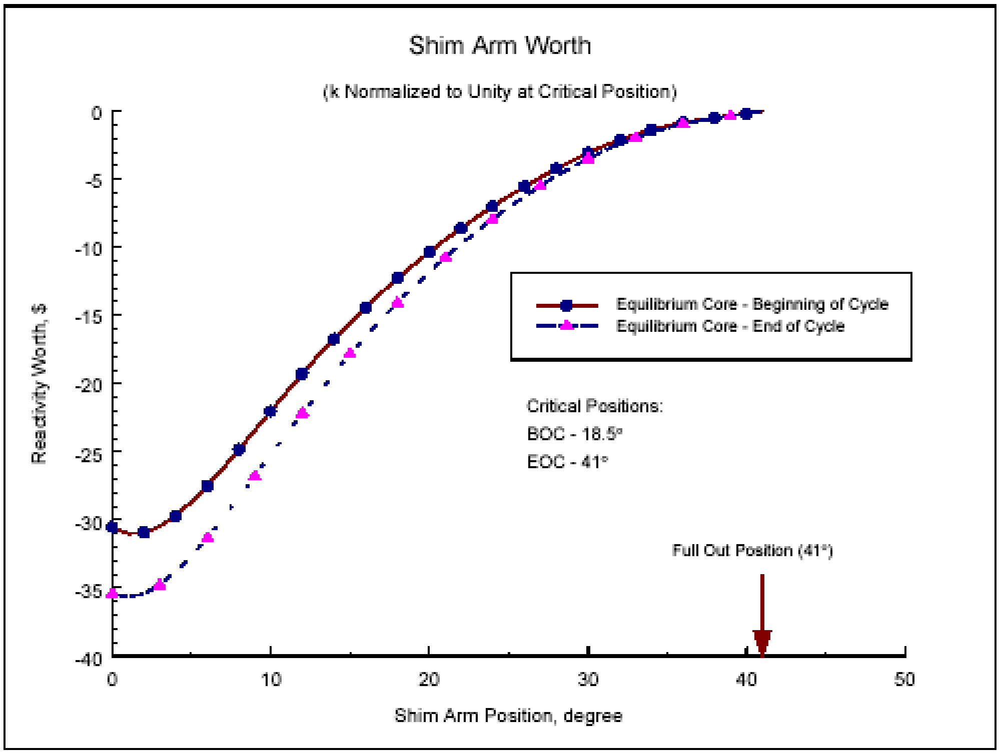


**Figure 13.1: Shim Safety Arm Reactivity Worth As A Function Of Angular Position At EOC**

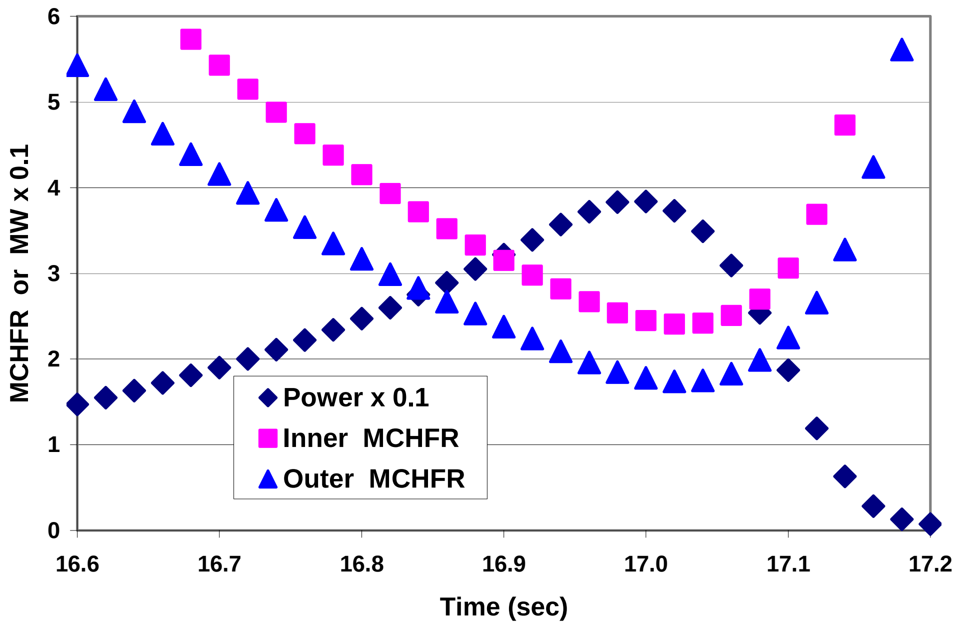


**Figure 13.2: Startup Accident (EOC)**
(Reactor Power (x 0.1) and Minimum Critical Heat Flux Ratios for the Inner and Outer Plenums)

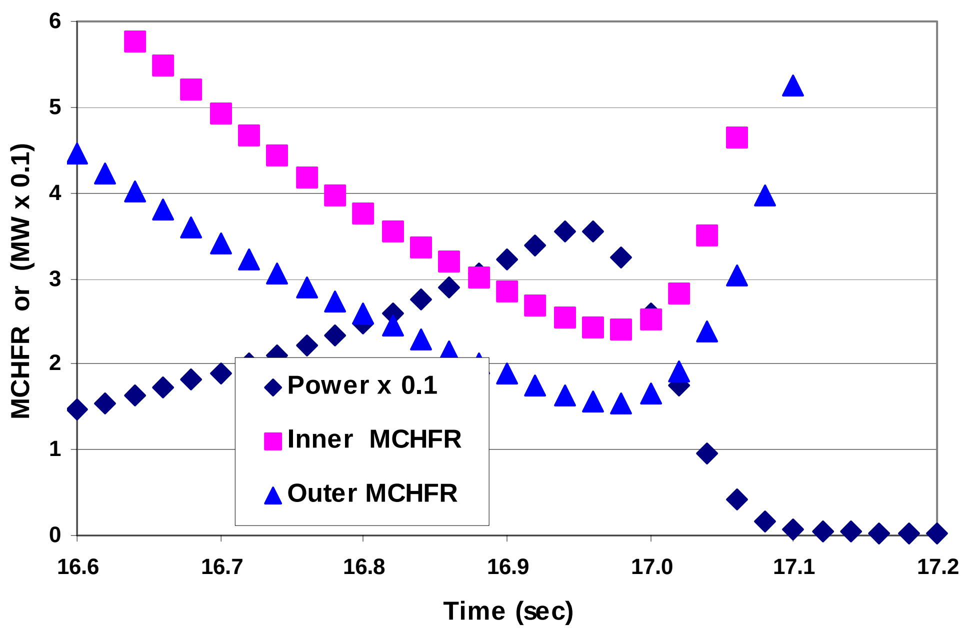


**Figure 13.3: Startup Accident (SU Core)**
(Reactor Power (x 0.1) and Minimum Critical Heat Flux Ratios for the Inner and Outer Plenums)

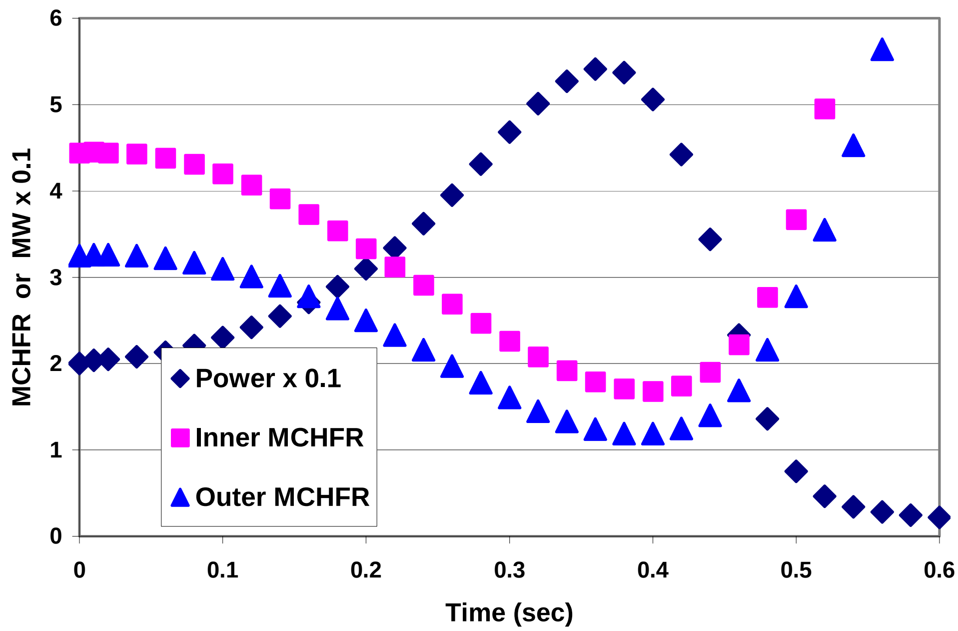


**Figure 13.4: Maximum Reactivity Insertion (EOC)**
(Reactor Power (x 0.1) and Minimum Critical Heat Flux Ratios for the Inner and Outer Plenums)

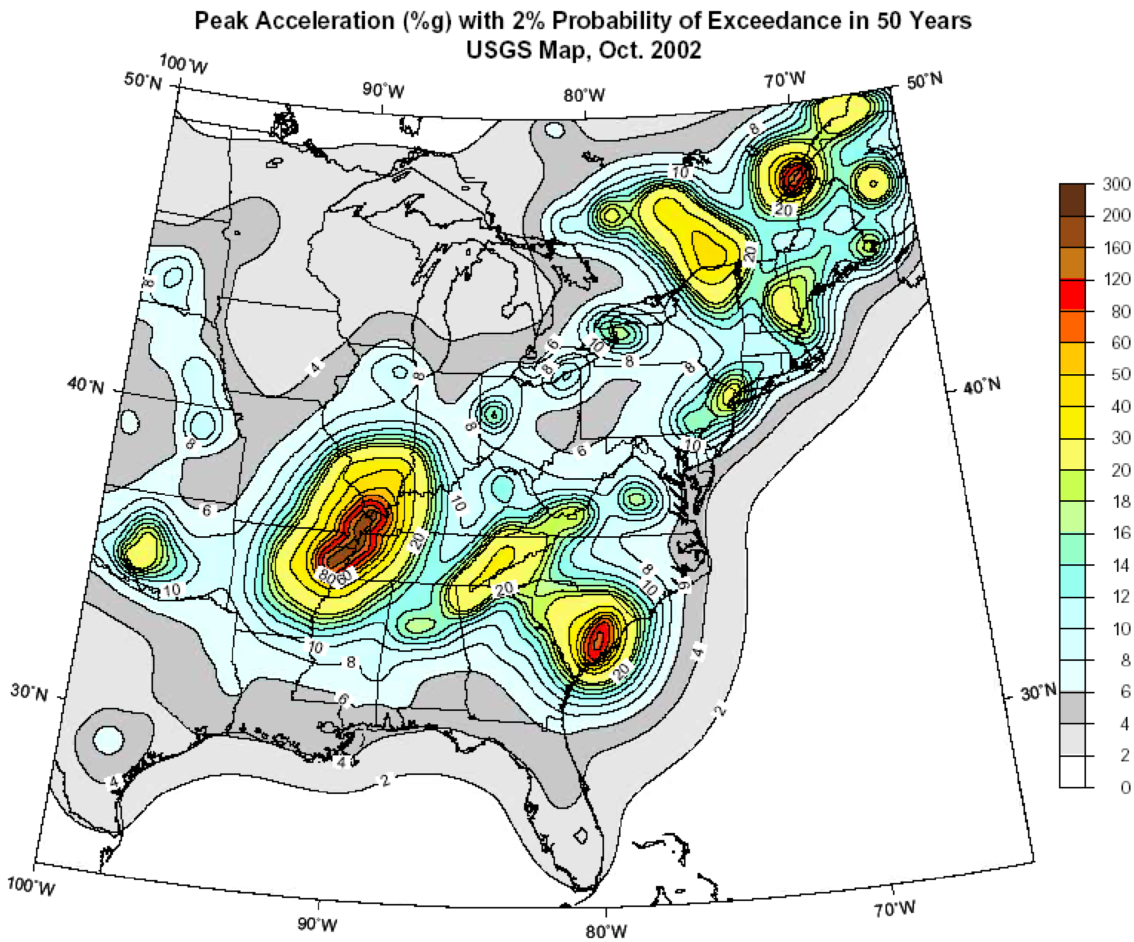


**Figure 13.5: USGS 2002 National Seismic Hazard Maps, Central and Eastern US maps**