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Criticality Risks During Transportation of Spent Nuclear Fuel

Revision 1

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Final Report, December 2008

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REPORT SUMMARY

This report presents a best-estimate probabilistic risk assessment (PRA) to quantify the frequency of criticality accidents during railroad transportation of spent nuclear fuel casks. The assessment is of sufficient detail to enable full scrutiny of the model logic and the basis for each quantitative parameter contributing to criticality accident scenario frequencies. The report takes into account the results of a 2007 peer review of the initial version of this probabilistic risk assessment, which was published as EPRI Technical Report 1013449 in December 2006.

Background

Risks during transportation of spent nuclear fuel have been addressed in two studies, NUREG/CR-4829, *Shipping Container Response to Severe Highway and Railway Accident Conditions*, and NUREG/CR-6672, *Reexamination of Spent Fuel Shipment Risk Estimates*. These studies assess risks of accidents capable of breaching the transportation package and resulting in the release of radioactive material to the environment, but do not quantify the frequency of criticality accidents directly.

Objectives

To provide a realistic perspective of criticality risks during railroad transportation of spent nuclear fuel casks for consideration by decision-makers imposing administrative and design controls on spent-fuel dry storage and transportation casks.

Approach

The research team examined the risks of criticality during railroad transportation by evaluating each event necessary for criticality. The evaluation uses a combination of existing reports, new assessments, and engineering judgments to quantify each potential contributor to a criticality event.

Results

The likelihood of any accident that has a potential for criticality over a total of 11,000 shipments is estimated to be below $2x10^{-12}$, which constitutes a negligible risk. This result arises from a number of independent factors:

- The extremely low likelihood that a railroad accident will produce the damage and immersion needed to achieve criticality, as determined by U.S. NRC-sponsored research
- The wide safety margin for the effective multiplication factor, k_{eff}, when fuel is loaded into a spent-fuel cask in accordance with its Certificate of Compliance
- The very low likelihood of an error in the recorded burnup of fuel assemblies due to flux mapping measurements and the use of fuel assembly burnup to predict and verify core

performance during active fuel cycles in the core. Fuel assembly burnup is accurately measured, tracked, and used to control reactivity during power generation. The predictions provided by the core follow software are verified by the agreement between predictions of core reactivity and actual measurement of the core neutron flux during subsequent fuel cycles.

- The very low likelihood of a misload due to the controls and verification requirements followed when loading fuel assemblies into the spent-fuel cask
- The ability to access core burnup and Special Nuclear Material accountability data at any time prior to shipment of a spent fuel cask offsite in order to verify compliance to the cask's Certificate of Compliance
- The fact that criticality may not be achieved even if the cask has been misloaded, damaged, and immersed in water. The misload must introduce a significant additional amount of reactivity, and the damage must be such that water can leak into the cask to allow the moderation needed for criticality.

Given these results, requiring a measurement that confirms the reactor record of assembly burnup produces negligible reductions in criticality risks.

EPRI Perspective

The objectives of the EPRI program are to support U.S. industry goals defined as follows: (1) the overarching goal is regulatory acceptance of a burnup credit methodology that takes credit for the negative reactivity that is practical including all fissile actinides, most neutron absorbing actinides, and a subset of fission products that account for the majority of the available credit from all fission products; (2) the methodology should be applicable to both pressurized water reactor (PWR) and boiling water reactor (BWR) spent fuel; and (3) unnecessary operational burdens imposed on plant personnel involved in loading spent fuel in burnup-credit-approved dry storage and transportation systems should be eliminated.

Using risk-informed arguments coupled with cost-benefit analyses demonstrates that use of burnup-credit-designed dry storage and transportation systems should be encouraged to minimize radiological and non-radiological risks. Favorable cost-benefit analysis results mostly derive from a reduction in the number of shipments, while enhancements in safety mostly derive from reducing handling and non-radiological transportation risks.

Keywords

Risk assessment
Burnup credit
Criticality safety
Spent fuel
Spent-fuel transportation
Misloading
Railroad transportation

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PROJECT SUMMARY

1.1 Objective

The objective of this study is to perform a probabilistic risk assessment (PRA) to quantify the frequency of criticality accidents during railroad transportation of spent nuclear fuel casks that take credit for burnup. The risk assessment evaluates the likelihood that errors in fuel selection and/or fuel handling would result in a spent fuel cask that would not conform to the Certificate of Compliance (CoC) for that cask. As a result of such errors, a criticality event could occur if the cask is breached and submerged in water following an accident during transportation of the cask. The study also updates previous estimates of the rate of railroad accidents capable of producing these conditions. It uses these evaluations with results from existing U.S. Nuclear Regulatory Commission (NRC) and EPRI reports to estimate the frequency of criticality accidents during railroad transport. The assessment is of sufficient detail to enable full scrutiny of the model logic and the basis for each quantitative parameter contributing to the criticality accident scenario frequencies.

1.2 Background

In order to qualify for independent storage, 10 CFR 72.236 requires that each model of cask be designed to assure that it to meets explicit safety criteria for subcriticality, shielding, confinement, and heat removal. Following a review of the cask's safety analysis against these criteria, the NRC approves a Certificate of Compliance that explicitly states the type and condition of fuel that may be safely loaded, including initial enrichment and burnup. The criticality analysis utilizes the conditions specified in 10 CFR 71.55(b) to define the criteria under which it is done. It is a regulatory requirement of 10 CFR 71.55(b) that

> "a package used for the shipment of fissile material must be so designed and constructed and its contents so limited that it would be subcritical if water were to leak into the containment system, or liquid contents were to leak out of the containment system so that, under the following conditions, maximum reactivity of the fissile material would be attained:
>
> (1) The most reactive credible configuration consistent with the chemical and physical form of the material;
>
> (2) Moderation by water to the most reactive credible extent; and

> (3) Close full reflection of the containment system by water on all sides, or such greater reflection of the containment system as may additionally be provided by the surrounding material of the packaging."

NUREG-1617, *Standard Review Plan for Transportation Packages for Spent Nuclear Fuel*, provides the existing recommendations for the NRC staff to proceed with acceptance, on a cask-specific basis, of a burnup credit approach in the criticality safety analysis of pressurized water reactor (PWR) spent fuel casks. ISG-8 Revision 2, *Burnup Credit in the Criticality Safety Analysis of PWR Spent Fuel in Transport and Storage Casks*, provides additional recommendations for ensuring subcriticality, one being that a measurement confirms the reactor record of the burnup of each assembly loaded into the cask. Specifically, Recommendation 5: *Assigned Burnup Loading Value*, states:

> "Administrative procedures should be established to ensure that the cask will be loaded with fuel that is within the specifications of the approved contents. The administrative procedures should include a measurement that confirms the reactor record for each assembly. Procedures that confirm the reactor records using measurement of a sampling of the fuel assemblies will be considered if a database of measured data is provided to justify the adequacy of the procedure in comparison to procedures that measures each assembly.
>
> The measurement technique may be calibrated to the reactor records for a representative set of assemblies. For confirmation of assembly reactor burnup record(s), the measurement should provide agreement within a 95 percent confidence interval based on the measurement uncertainty. The assembly burnup value to be used for loading acceptance (termed the assigned burnup loading value) should be the confirmed reactor record value as adjusted by reducing the record value by a combination of the uncertainties in the record value and the measurement."

This study also examines the risk benefit of requiring an independent measurement during loading of a spent fuel assembly into the spent fuel cask to confirm the reactor record of the burnup of each assembly loaded into a cask for which burnup credit is taken.

1.3 Review of Published Research

1.3.1 NRC Sponsored Research on Spent Fuel Shipment Risk

Risks during transportation of spent nuclear fuel have been addressed in two studies, NUREG/CR-4829, *Shipping Container Response to Severe Highway and Railway Accident Conditions*, (Reference 1), also known as the "Modal Study," and NUREG/CR-6672, *Reexamination of Spent Fuel Shipment Risk Estimates*, (Reference 2). These studies focus on severe accidents capable of breaching the transportation package, resulting in the release of radioactive material to the environment.

For railroad transportation, the studies use data and accident reports collected by the Federal Railroad Administration (FRA) to calculate railroad accident rate. It accounts for all types of accidents involving both passenger and freight trains, on all classes of track, at all speeds. The studies then estimate the conditional likelihood of accidents and loading conditions that could cause cask damage; the likelihood of the damage will lead to radiation releases from an accident involving a train carrying spent fuel casks.

An accident that could produce some of the conditions necessary for a criticality event to occur is an extremely small subset of the train accident rate calculated using the FRA data. The only significant reference to the physical conditions that could produce criticality is given on pages 9-24 and 9-25 of the Modal Study, which states:

> "For large casks containing more than three PWR bundles, the effectiveness of measures used to prevent a criticality event can be reduced under extreme loading conditions. Any reduction in criticality safety depends on both the cask and fuel basket design. However, since the margins used to prevent criticality are very high, and since careful evaluations of the criticality analysis and the design features are performed during cask licensing, the possibility of a criticality event is small even under extreme loading conditions. Using probabilistic methods in Section 5.0, the probability of a rail cask's having a structural response greater that 2% strain (S_2) and becoming submerged in water is estimated to be 0.00000078%, given an accident. Using the accident and rail shipment rates in this study, this type of accident estimated to occur approximately once every ten million years."

The probability given in the above quotation, $7.8x10^{-9}$, is not an estimate of the likelihood that all the conditions stated in 10 CFR 71.55(b) will fully occur. It is an estimate of the likelihood that a rail cask is subjected to an accident that can breach the cask containment at a location where water can leak into the cask should the breach occur.

The calculations that support the estimate were not published and are no longer available. However, the methodology described in NUREG/CR-4829 and its appendices provide some insight into what is required to produce the conditions described above. The most important insight is that the baseline accident rate to which this factor was applied encompassed the following conditions:

- All trains, which include passenger and work trains, as well as freight trains
- All grades of track, which include local and switchyard track as well as high grade main line track
- At all speeds
- During the period from 1976-1984, which does not account for recent history

The limiting conditions listed below indicate that the conditional likelihood of $7.8x10^{-9}$ per train accident corresponds to the special set of accident conditions listed below.

Transport by freight train.	At the time the Modal Study was accomplished, FRA data did not differentiate between freight transport and other forms of rail activity. Most recent FRA data summarized in Section 3 of this report indicates that freight train is slightly lower than that of a general population of all train traffic.

Control of train and traffic	Due to the weight and hazardous nature of the spent fuel, shipments will be controlled at least as closely as HAZMAT[1] shipments. Section 3 of this report indicates that trains carrying HAZMAT have accident rates below that of generic freight trains.
Accidents producing an impact	Only selected accidents have the potential to produce an impact load on a cask. These include only collisions and derailments.
The threshold for damage to a cask	The threshold for damage would require an impact at a relative speed of at least 60 mph. As shown in Section 3, FRA accident records indicate that over 98% of accidents occur at train speeds of 60 mph or less.
Direct impact on a hard object	A hard object must be present at the point of impact, and the impact force must be directed to the cask. In order for sufficient impact energy to be applied to the cask, the impacting object must bypass the impact limiter and be of sufficient hardness to not absorb a significant fraction of the energy itself.
Configuration of impact	The impact must occur in a manner to cause a deformation sufficient to fracture the containment. An impact over a broad area would allow a significant part of cask to absorb the energy or transfer it into kinetic energy in the form of a rebound.
Proximity to water	To become susceptible to criticality, the cask must end up submerged in water. This can only happen where the rail is either over or near water deep enough allow this to happen.
Sequence of events	Because of the energy absorbing capability of water, the cask must impact the object that would produce the damage before entering the water.

Considering the above factors, the conditional likelihood of 7.8×10^{-9} per train accident used in the Modal Study is judged to be realistic.

1.3.2 EPRI Sponsored Research on Spent Fuel Burnup and Criticality

EPRI has addressed the issue of criticality safety while taking credit for burnup by examining both the uncertainty in the residual reactivity in spent fuel and the impact of introducing "under-burned" fuel into a waste package. EPRI 1003418 (Reference 3), *Burnup Credit–Technical Basis for Spent-Fuel Burnup Verification*, examined the impact of misloading both "under-burned" and fresh fuel into a 24-PWR-assembly cask design containing fixed boron neutron absorbers that meet only the criteria for storage in an independent spent fuel storage installation (ISFSI), but

[1] HAZardous MATerial

cannot be shipped under current regulations. These calculations accounted for actinide depletion and buildup of five neutron-absorbing fission products. The results are shown in Figure 1-1, for spent fuel with a discharge burnup of 45 GWd/MTU from a hypothetical fuel cycle utilizing 5% enriched fuel. The value for k_{eff} of 0.88 with no misloaded fuel assemblies indicates that there is considerable margin for uncertainty in the calculations when the presence of neutron absorbers actually in the fuel is accounted for.

The two sensitivity cases in Figure 1-1 show the impact of the substitution of up to eight assemblies (i) with a burnup of only 25 GWd/MTU (about 56% of the design burnup), and (ii) consisting of fresh fuel. For both cases, the substitution of these assemblies is made by grouping them together in the middle of the cask, which produces the highest k_{eff}. It can be seen in the figure that the substitution of up to eight fuel assemblies with a burnup of 25 GWd/MTU results in a maximum k_{eff} of only 0.95. Furthermore, it shows that more than one assembly of fresh fuel must be misloaded into the cask to result in a k_{eff} greater than 0.95, while three are needed to produce criticality.

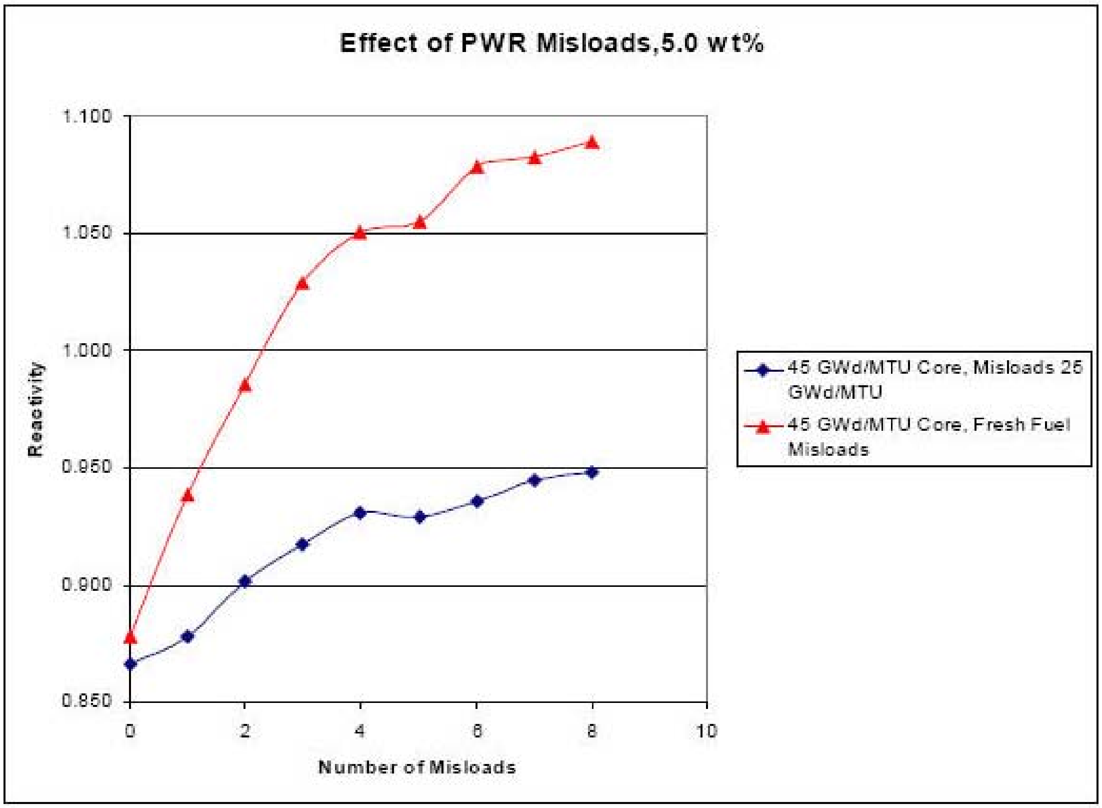

Figure 1-1
Effect of Misloaded Fuel Assemblies on the k_{eff} of a Conceptual 24-PWR Spent Fuel Cask

Discussions with a number of refueling engineers at various utilities have indicated that there is a distinct difference in the appearance of fresh and once-burned fuel assemblies, as illustrated in Figure 1-2. This figure shows an arrangement of fuel assemblies during a refueling operation. The fresh fuel assemblies have their original metallic color, while the once-burned assemblies have been darkened by corrosion. This risk assessment takes no credit for the ability of members of the refueling team to recognize the differences between a fresh fuel assembly and a once-burned fuel assembly. However, the readily recognizable difference in appearance provides additional assurance that the likelihood of a misload developed in this assessment will not involve a fresh fuel assembly.

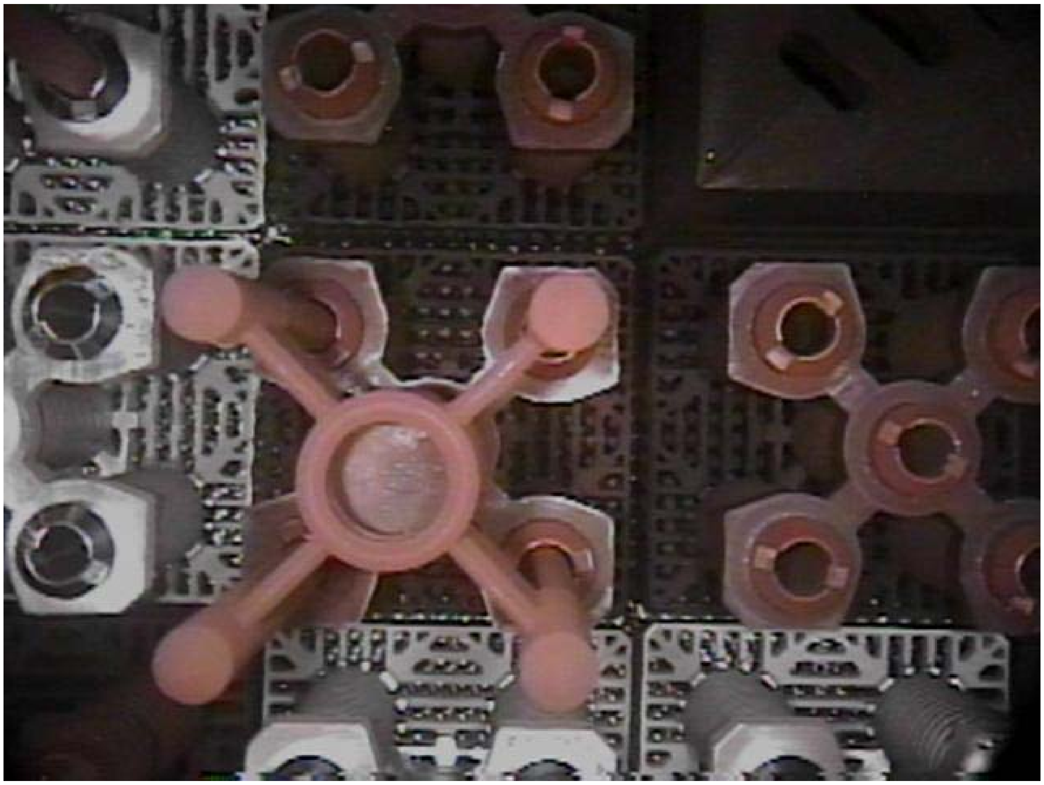

Figure 1-2
New and Once-Burned Fuel in a Reactor Core (Upper right is the reactor pressure vessel)

A sensitivity case documented in EPRI 1003418 shows that cask size does not significantly alter the k_{eff} of the package, as shown in Figure 1-3. For the calculations using spent fuel with the same total burnup, the criticality calculations for a 32 assembly cask produce essentially the same k_{eff} as a 24 assembly cask. This results from the fact that the calculation groups all the under-burned assemblies in the center of the cask in their most reactive configuration. The misloaded fuel acts like a small core surrounded by a reflector of the more highly burned assemblies directly adjacent to the misloaded fuel. The addition of assemblies beyond the adjacent assemblies increases k_{eff} only to the extent that net neutron leakage out of the misloaded region is reduced. As the immediately adjacent assemblies provide the vast majority of reflection back into the misloaded group, the overall effect on k_{eff} of using a larger cask is insignificant.

EPRI 1015050, *Fuel Relocation Effect for Transportation Packages*, (Reference 3) evaluated fuel relocation effects on criticality following accidents that produce severe damage to transportation packages. The results of this study stated:

> "Application of the Double Contingency Principle to unlikely, but credible, "worst-case" scenarios shows that the maximum reasonable reactivity increase is either less than the administrative margin of 0.05 for scenarios involving physical changes to fuel assembly rod arrays or more likely to result in a substantial reactivity decrease for scenarios involving fuel pellet arrays."

This result indicates that the criticality calculations conducted to support the application for a cask design Certificate of Compliance will remain valid for cask damaged in a transportation accident.

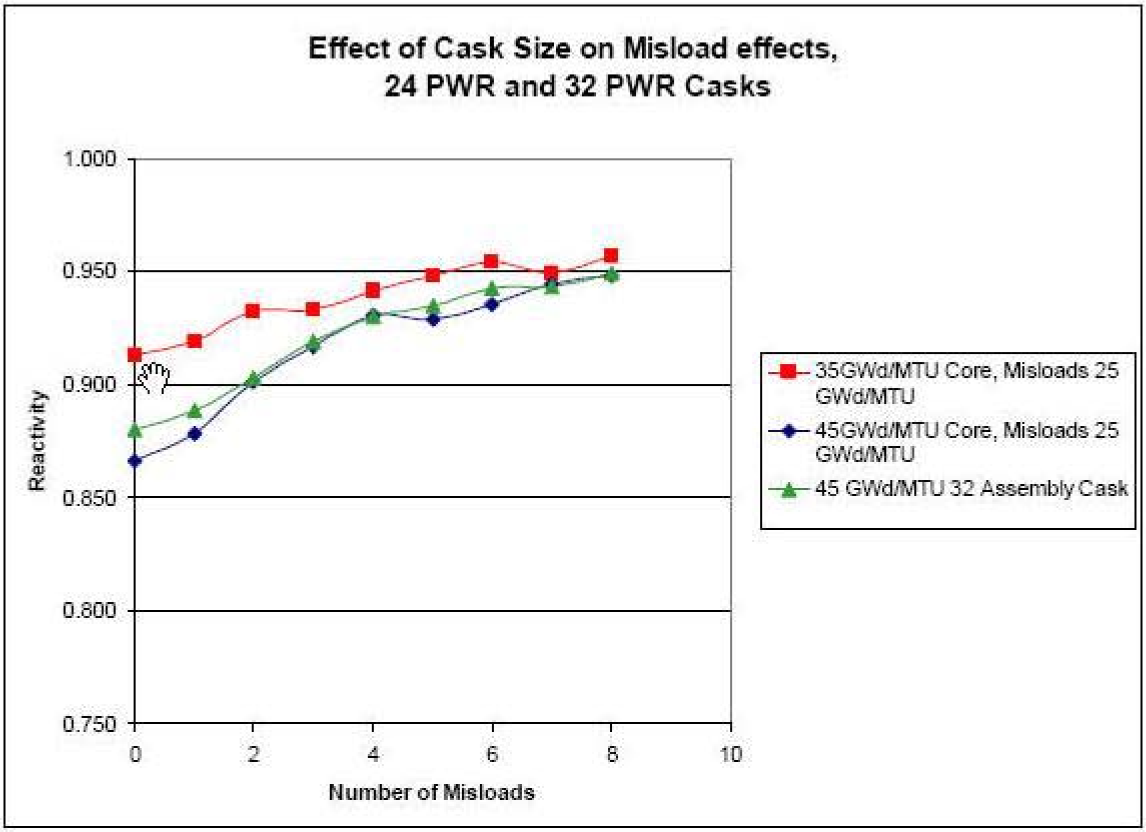

Figure 1-3
Effect of Cask Size on Misload Effects, 24 PWR and 32 PWR Casks

1.4 Sequence of Events Needed for Criticality

The criticality evaluations required by 10 CFR 72.236 provide high confidence that a spent fuel cask will remain subcritical under optimum moderation and reflection conditions if the cask is loaded in conformance with the Certificate of Compliance approved by the NRC for that cask design. Therefore, the cask must be misloaded in order to be susceptible to criticality. The risk assessment conservatively assumes that if a single fuel assembly is misloaded for any reason, dependencies between errors will result in additional misloaded assemblies, leading to a worst case configuration that would be susceptible to criticality if the cask is breached and submerged in water due to an accident during transportation.

Figure 1-4 illustrates the sequence of events that would lead to a criticality event, with the undesired pivotal event resulting in the downward branch. The following paragraphs summarize each of these events.

Track and Record Burnup by Fuel Assembly (F/A) Serial Numbers (SN) during Fuel Cycles

Spent fuel assemblies are selected based on their initial enrichment, burnup within the reactor, and cooling time following discharge. Two errors could result in misidentified assemblies being on the fuel selection document: 1) an error in the reactor measurements and/or software that track the burnup of each fuel assembly, or 2) an error during the selection process resulting in assemblies not conforming to the cask's Certificate of Compliance being recorded in the fuel selection document. The sources of error and safeguards to prevent these errors are evaluated in Section 2.1.

Load a Spent Fuel Cask (SFC) with F/As that Meet its Certificate of Compliance

In this assessment, a total of 32 spent PWR fuel assemblies are assumed to be loaded, because casks have been manufactured with 32 positions. The use of 32 fuel assemblies results in a conservative estimate for the likelihood of a misload for PWR spent fuel casks. The plant's fuel handlers load spent fuel casks following approved plant procedures. A qualified crew performs these activities as a routine dedicated activity with no set time limit.

Special Nuclear Material (SNM) Inventory Verifications Detect Errors prior to Shipment

Once loaded, a SFC may remain at the plant's Independent Spent Fuel Storage Installation (ISFSI) for years prior to shipment. At any time during the storage period, the records associated with that SFC may be accessed and checked against the SNM accountability database and the core burnup records. In addition, the transfer documents, video records, and annual inventories may be reviewed to verify the serial numbers of the fuel assemblies in each dry shielded cask (DSC) and the spent fuel pool (SFP).

Sections 2.2 through 2.4 assess the likelihood that the crew will misidentify at least one assembly during this process, and that the error will not be detected prior to shipment of the spent fuel cask. The estimate is based on a detailed evaluation of the procedures that have been used to load spent fuel for dry storage at an example plant.

Receive Fuel Assemblies at Plant	Track and Record Burnup by F/A SN during Fuel Cycles	Load a SFC in accordance with its Certificate of Compliance	SNM inventory verifications detect error prior to shipment	Accident during transport (2000 mi trip)	Cask damaged with > 2% strain <u>AND</u> submerged in water	End State
	Correct burnup assigned to F/A SNs in Central	SFC Loaded Correctly	N/A	N/A	N/A	No possibility of crticality
		Incorrect S/N(s) loaded	Misload Detected by verifications			SFC reevaluated/ repackaged
			SFC with incorrect S/N(s) Shipped	Load arrives safetly		No accident, no criticality
				Cask subjected to accident conditions	No moderation	No moderation, no criticality
					Conditions required for criticality	Accident with potential for criticality
	Incorrect burnup assigned to F/A SN	Incorrect S/N(s) loaded	Misload Detected by verifications			SFC reevaluated/ repackaged
			SFC with incorrect S/N(s) Shipped	Load arrives safetly		No accident, no criticality
				Cask subjected to accident conditions	No moderation	No moderation, no criticality
					Conditions required for criticality	Accident with potential for criticality

Figure 1-4
Sequence of Events Necessary for Railroad Transport of Spent Nuclear Fuel to Produce a Potential Criticality Event

Train Transporting the Cask Has an Accident

NUREG/CR-4829 and NUREG/CR-6672 used experience and accident data from the Federal Railroad Administration (FRA) to estimate the frequency of all reportable train accidents (freight, passenger, and working) at all speeds and under all operating conditions as the starting point of its damage assessment. Although spent fuel shipments will be made on freight trains, operational freight train-mileage has been reported in such a way that it can be effectively separated from other train operational data only since January 1997. Therefore, the estimate of train accident rates in earlier studies included all accident events in order to keep the numerator and denominator of the accident rate equation consistent. NUREG/CR-4872 (p. 2-3) stated the basis for this assumption. "Because over 90% of all train mileage is attributed to freight trains, there is no significant difference in applying data based on all trains to freight trains in order to estimate accident rates, accident velocities, fire frequencies, etc."

Section 3 of this report updates that frequency to reflect the most recent railroad experience using the online database maintained by the Federal Railroad Administration. The data in the online database confirm the accuracy of the accidents rates reported in NUREG/CR-4829 and NUREG/CR-6672 and update the accident rate to reflect experience since January 2000. FRA data fields are now available (e.g., freight and passenger train-miles) to assist the estimate of the frequency of freight train accidents that would be more closely applicable to spent fuel transportation. In addition, although not used in the risk estimate, Section 3 calculates a set of accident rates for freight trains carrying HAZOP[2] materials to provide some insight into train controls that can reduce the likelihood of the conditions needed for criticality.

Cask Has a Structural Response > 2% Strain and Becomes Submerged in Water

Both of these events require extensive evaluation of the circumstances of a railroad accident, which was conducted in NUREG/CR-4829. That document states on p. 9-24 and 9-25, "Using probabilistic methods in Section 5.0, the probability of a rail cask's having a structural response greater that 2% strain (S_2) and becoming submerged in water is estimated to be 0.00000078%, given an accident." The percentage stated in the quote converts to a conditional likelihood of 7.8E-09/accident. It is assumed in this report that this value is valid.

The accident to which this quote refers is the overall frequency of all reportable train accidents at all speeds and under all operating conditions. However, based on the assumption stated above that "there is no significant difference in applying data based on all trains to freight trains in order to estimate accident rates, accident velocities, fire frequencies, etc," this factor can also be applied to the freight train accident rate since January 2000.

Water Leaks into the Cask, Producing Sufficient Moderation and Reflection

The damage to the cask must be oriented such that water can leak into the spent fuel cask. For this to occur, the damage must extend to the upper shell of the cask as it lays in the water to

[2] HAZard and OPerability

allow an escape path for the inert gases in the cask. If no such damage and orientation occurs, the trapped gases will prevent full flooding of the fuel assemblies.

The likelihood of this event is not quantified, and it is conservatively assumed in this risk assessment that any strain greater than 2% will produce a breach in the cask and that water will leak into the cask producing optimum moderation and reflection conditions.

<u>The Misloaded Fuel Has Sufficient Reactivity to Achieve Criticality</u>

In this assessment, it is assumed that any error in either the reactor records or in the preparation of the fuel selection document from the reactor records will result in sufficient underburned fuel assemblies being loaded in the spent fuel cask to achieve criticality should optimum moderation and reflection be achieved. Consequently, this event is assumed to occur.

This is an extremely conservative assumption and is not included in the event tree. As discussed in Section 1.2.2, EPRI 1003418 (Reference 3) has shown that there is considerable reactivity margin to account for uncertainty and misloading error in the calculations. The two sensitivity cases shown in Figure 1-1 indicate that it is extremely unlikely a criticality event will occur, unless fresh fuel assemblies are misloaded as spent fuel. Furthermore, since it is a practical goal of utilities to make maximum use of the fuel it purchases, it is highly unlikely that misidentified fuel will have significantly lower burnup. Because every reactor will have a set of spent fuel that reflects its own past fuel management policies, the likelihood that criticality will not be achieved despite the misload cannot be quantified straightforwardly.

1.5 Results

1.5.1 Fuel Selection and Loading Error Rates

The event tree in Figure 2-5 summarizes the evaluation of the spent fuel selection and cask loading procedures, as well as accounting for the opportunity to review the entire history of each fuel assembly to detect burnup, selection, or loading errors prior to shipment. It indicates that the total likelihood of a spent fuel cask shipment with one or more misloaded fuel assemblies is 2.6E-06 per cask loaded, assuming procedures are in place to verify the identity and burnup of the fuel assemblies against core burnup and SNM accountability data prior to shipment.

1.5.2 Railroad Accident Rates

Table 1-1 summarizes the evaluation of train accident rates used to quantify the criticality risk assessment. The detailed data used to calculate these rates are given in Section 3.4. The table shows that railroad accident rates have improved relative to the earlier studies. The trend results from the significant reduction in accident rates over the ten-year period from 1978 to 1988, as shown in Figure 3-4.

The event tree uses the accident rate based on 2000-May 2006 freight train accidents and miles for the pivotal event, as the Modal Study used the more general accident rate only because experience was not broken down by type of train prior to 1997. The more recent data allow the freight train accident rate to be calculated directly from FRA data. This direct calculation indicates that the earlier assumption that freight train rates are not significantly different from the overall accident rate was conservative, but reasonable. The lower accident rate for freight trains is plausible, since freight trains are employed primarily for long haul transportation, while passenger and special purpose trains tend to operate over shorter distances and in more populated and congested areas.

Table 1-1
Summary of Railroad Accident Rates Used for the Criticality Risk Assessment

Source Document	Data Time Period	Reported Accident Rate (Events/Train-Mile)	Extracted from FRA On-Line Database		
			Accident Rate (Events/Train-Mile)	Number of Reportable Accidents	Total Railroad Train-Miles
NUREG/CR-4829 (Modal Study)	1975-1982	1.19E-05	1.19E-05	68,489	5,762,577,462
NUREG/CR-6672	1988-1995	4.57E-06	4.57E-06	22,594	4,948,189,617
This Report	2000-May 2006		4.33E-06	20,779	4,803,036,079
This Report (Freight Train Accidents and Mileage)	2000-May 2006		2.67E-06	9,697	3,635,445,944

1.5.3 Quantification of Criticality Risks During Transportation

Table 1-2 combines the assessments conducted in this study to estimate the risk associated with future spent fuel shipments. The number of anticipated shipments and years is taken from NUREG/BR-0292, *Safety in Spent Fuel Transportation* (Reference 5). The shipping distance is a rough but reasonable estimate considering the uncertainties of the some of the parameters in the risk assessment.

Table 1-2
Summary of the Risk of Criticality during Railroad Transportation

Description	All Trains	Freight Trains
Train Accidents per Train-Mile (All Accidents, All Speeds, All Track Classes), 2000 - May 2006.	4.3E-06	2.7E-06
Probability of Accident of Interest, Given Any Accident (>2% Strain and Immersion) per Modal Study	7.8E-09	7.8E-09
Frequency of Accidents of Interest for Criticality/Train-Mile	3.4E-14	2.1E-14
Assumed Average Number of Miles per Shipment	2,000	2,000
Frequency of Accidents of Interest for Criticality/Shipment	6.8E-11	4.2E-11
Likelihood of Shipping a Misloaded Spent Fuel Cask	2.6E-06	2.6E-06
Likelihood of an Accident with a Potential for Criticality/Shipment	1.8E-16	1.1E-16
Number of Shipments per Year = 11,000/24 Years (from NUREG/BR-0292, Reference 13)	458	458
Frequency of Potential Criticality Accidents per Year	8.1E-14	5.0E-14
Likelihood of a Potential Criticality Accident over all 11,000 Shipments	1.9E-12	1.2E-12

1.6 Discussion

Table 1-2 shows that the likelihood of a criticality event during the shipment of spent fuel is negligible. The likelihood of any accident that has a potential for criticality over a total of 11,000 shipments is estimated to be below $2\text{x}10^{-12}$, which is negligible. This result arises from a number of independent factors:

- The extremely low likelihood that a railroad accident will produce the damage and immersion needed to achieve criticality, as determined by the U.S. NRC-sponsored research presented in Section 1.2.1.
- The very low likelihood of an error in the recorded burnup of fuel assemblies due to flux mapping measurements and use of fuel assembly burnup to predict and verify core performance during active fuel cycles in the core.

- The low likelihood of a misload due to the controls and verification requirements followed when loading fuel assemblies into the spent-fuel cask.
- The ability to access core burnup and Special Nuclear Material accountability data at any time prior to shipment of a spent fuel cask offsite in order to verify compliance to the cask's Certificate of Compliance.

An important factor in achieving the negligibly small risk is the ability to access core burnup and Special Nuclear Material accountability data at any time prior to shipment of a spent fuel cask offsite in order to verify compliance to the cask's Certificate of Compliance. As discussed in Section 2.1, 10 CFR 74.19, *Recordkeeping*, subparagraph (a) requires that licensees keep records showing the receipt, inventory (including location and unique identity), acquisition, transfer, and disposal of all special nuclear material in its possession regardless of its origin or method of acquisition until at least 3 years following transfer or disposal of the material, and the January 2007 draft of ANSI 15-8, *Special Nuclear Material Control and Accounting Systems for Nuclear Power Plants*, provides reasonable best practice guidelines for a practical and effective SNM system. It is strongly recommended that utilities institute procedures that utilize these resources to verify the content of their dry spent fuel casks prior to shipment.

The results of this risk assessment are the outcome of the propagation of point estimates that are judged to provide decision makers with reasonable and sufficient evidence to conclude that the risk of a criticality accident during rail transportation is negligibly small. In some cases (e.g., the value of 7.8E-09 cited for probability of full immersion and strain damage >2% given a train accident), the detailed basis and statistical nature of the parameters were not found in the source documents. In other cases, the individual estimates were made with engineering judgment that allowed for a wide margin of safety (e.g., assuming that any misload could make a SNF Cask susceptible to criticality despite the conclusions of EPRI 1003418 (Reference 4) that misload of fresh fuel is needed for that to occur). A more detailed evaluation of the uncertainties could provide decision-makers with a best-estimate mean, median, or bounding value of the risk, but in the opinion of the analysts, the conclusions of the current analysis would not change.

The research cited in Section 1.2 and the assessments accomplished in Sections 2 and 3 of this report indicate that the above estimate of the negligible likelihood of a potential criticality accident could be lowered even more if the following items were accounted for:

- The risk assessment's use of the factor cited in the Modal Study of 7.8E-09 assumes that full moderation and reflection of the fuel would occur if the cask were damaged and submerged. In order for water to leak into the cask, the breach path would have to be oriented on the high side of cask in the water to allow the inert gas to leak out and sufficient water to leak in to achieve the moderation and reflection needed achieve criticality.
- Criticality calculations for establishing initial enrichment and burnup limits for a cask design's Certificate of Compliance (CoC) assume pure water as a moderator. The water that would surround and leak into a damaged cask following an accident will have contaminants that could absorb neutrons.
- Criticality calculations for establishing initial enrichment and burnup limits for a cask design's CoC assume that all spent fuel assemblies are at the burnup limit that would achieve

maximum reactivity. While some fuel assemblies will be at that limit, it is likely that they will have higher burnup than the limit, yielding additional margin against a criticality event.

- The risk assessment assumed that a cask having <u>any</u> misloaded fuel would be subject to a potential criticality event. However, the estimate of the human error rate for transferring incorrect assemblies into the cask (HATRN1) was largely based on data that included assemblies that only exceeded decay heat limits.
- Taking into account the actinides and a small, albeit significant, subset of fission products that are known to be produced during power operations, Figure 1-1 indicates that the substitution of fuel assemblies with a burnup of only 25 GWd/MTU (corresponding to a single burn of fuel assemblies designed for an active life of two fuel cycles) instead of 45 GWd/MTU into a cask designed only for ISFSI storage of 45GWd/MTU) results in a k_{eff} that levels at about 0.95. Based on this, one can infer that fresh fuel would have to be loaded into this cask in order for it to become susceptible to a criticality event following an accident during transportation.
- In general, assuming limiting conditions, a minimum of one fresh fuel assembly enriched at 5% is required to be in the center portion of the cask for producing a criticality event following an accident during transportation. However, as shown in Figure 1-2, fresh fuel can be easily distinguished from fuel that has been irradiated in a previous fuel cycle.

A number of operational safeguards and controls further reduce risks. For examples:

- The trains transporting spent nuclear fuel will be more closely controlled and monitored than the generic population of freight trains evaluated in the risk assessment. For example, train speed limits can be established below the 60 mph threshold speed needed to produce damage. They could be further reduced selectively for those stretches of track that have the close proximity to water deep enough to fully immerse a spent fuel cask.
- There is a very low likelihood that fresh fuel will be in the spent fuel pool when spent fuel casks are loaded. Since all fuel is handled by one group within a plant and spent fuel pool space is limited, spent fuel cask loading would typically be scheduled to be made early in a fuel cycle run to make room for the next refueling operation. The new fuel is received into the new fuel storage area, where it is inspected and stored to just prior to refueling. (Some plants that changed from a three to two cycle shuffle may have to transfer new fuel to the spent fuel pool to make room for the last shipments of new fuel, but the work necessary for processing of the new fuel will take priority over any loading of spent fuel casks.)
- Those few assemblies that do not undergo their full design burnup for some reason have been identified and should be well documented as exceptions. In addition, the coding of serial numbers on fuel assemblies can provide an indication of their history, which could also serve as a cue for recognizing a misloaded fuel assembly.

1.7 Conclusions

The risk of a criticality accident during the transportation of spent nuclear fuel is negligibly small and can be maintained that way with current utility fuel management and SNM accountability practices. This result arises from a number of independent factors:

- The accurate and verifiable methods by which fuel assembly burnup is utilized for management of the active core fuel cycle. The burnup of each fuel assembly is periodically updated by measuring the neutron flux throughout the core and used to predict and control core reactivity during each fuel cycle. The continued acceptable performance of the core provides validation that fuel assembly burnup updates are accurate.
- The very low likelihood of a misload of a misidentified fuel assembly due to the procedural controls and verification requirements followed when loading and shipping the spent fuel cask. The cask will not become critical if the fuel is loaded into a spent fuel cask per its Certificate of Compliance.
- The extremely low likelihood that a railroad accident will produce the damage and immersion needed to achieve criticality, as determined by NRC-sponsored research.

With the institution of procedures to make full use of the measurements and records fuel assembly burnup currently available to each licensee to verify the contents of each spent fuel cask prior to its shipment, an additional measurement that confirms the reactor record of burnup for each assembly when it is being loaded into the spent fuel will not produce a significant additional reduction of the already negligible probability of a criticality event during the transportation of spent nuclear fuel.

2
LIKELIHOOD OF SHIPPING A MISLOADED SPENT FUEL CASK

This section evaluates the likelihood of shipping a dual-purpose SFC that could be susceptible to criticality. This susceptibility is possible only if the SFC is misloaded with fuel assemblies that do not conform to the cask's Certificate of Compliance regarding initial enrichment and burnup. The section evaluates the procedures in place to control the utilization, movement, and storage of nuclear fuel within a typical nuclear plant. It then assesses the likelihood of human errors during the process that could produce a misload and the verifications that could reveal those errors before the SFC is shipped.

As the reactor physics codes that track burnup have been in use for a long period of time and are continually validated by neutron flux and heat generation measurements, it is assumed that these codes provide accurate results when properly used. Opportunities for human error can be associated with managing the fuel cycle, transferring information regarding the fuel assemblies, and selecting and loading fuel assemblies into a SFC. As these actions are not done in response to a reactor trip or other safety threatening initiating event, they do not require time-dependent diagnosis of an abnormal conditions or timely action. Consequently, they are analyzed as Type A actions under the guidelines of ASME RA-S-2002, *Standard for Probabilistic Risk Assessment for Nuclear Power Plant Applications*.

2.1 Control of Fuel Assemblies During Power Generation

The handling and tracking of fuel assemblies throughout their active life cycle in the plant provides the context under which the burnup of each fuel assembly is measured and recorded. As shown in the following paragraphs, fuel is handled and tracked by a combination of formal procedures containing independent verification, physical measurement of neutron flux and burnup calculations to support core fuel management, and the interface with a plant's Special Nuclear Material Control and Accounting System.

2.1.1 Receipt of New Fuel Assemblies

Figure 2-1 illustrates typical activities accomplished during the initial receipt and processing of fuel for a new cycle (designed Cycle N). Utilities purchase fuel assemblies for each fuel cycle based on licensing commitments and operational history. They purchase the number of fuel assemblies that they intend to load into the core for the next cycle. The manufacturer marks each fuel assembly with a serial number. Some manufacturers follow a systematic pattern for assigning fuel assembly serial numbers to assist the plant in controlling the loading pattern. For example, at one plant, the serial numbers have the following pattern:

X2R108

- X2 Identification of reactor unit
- R Fuel Cycle
- 1 Batch (0 to 5 based on the amount of burnable poison in the assemblies)
- 08 Index numbers of the assemblies within a batch

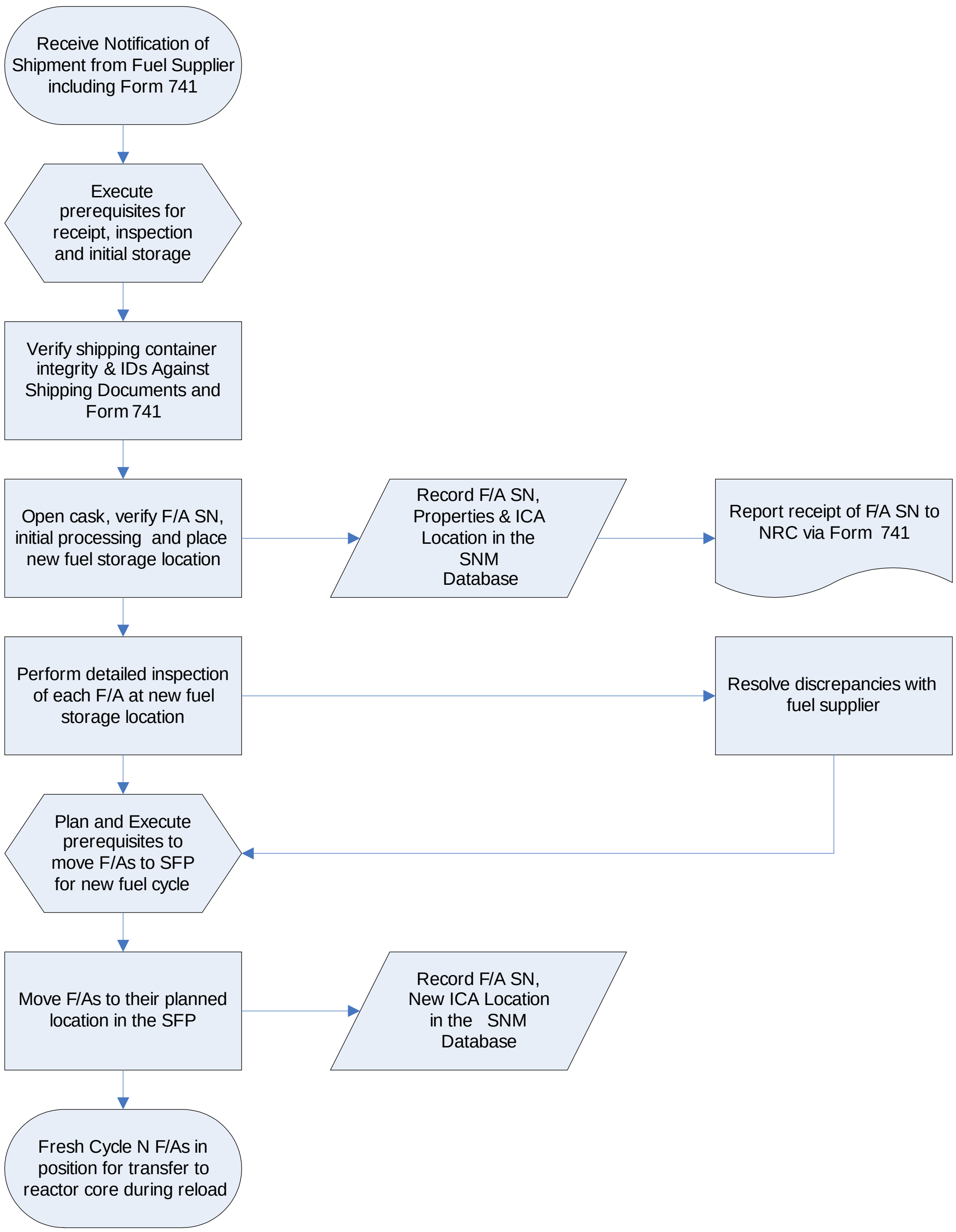

Figure 2-1
Typical Process for the Receipt of New Fuel at the Plant

New fuel is received and inspected by serial number in the new fuel storage area, where it can also be stored until refueling. Part of this process requires that the utility formally report receipt of the fuel to the NRC via the Form 741. Prior to refueling, it is transferred to a designated location in the SFP.

As the plant personnel physically inspect each fuel assembly, it is reasonable to assume that the likelihood of misidentification during receipt is negligible.

2.1.2 Control During Active Fuel Cycles

Unless it is damaged during a fuel cycle, each fuel assembly will participate in two or more fuel cycles prior to being stored in the SFP as spent fuel. Figure 2-2 illustrates the process for planning and executing a typical core reload for the Cycle N.

During Cycle N-1, plant in-core neutron flux measurements provide data to estimate the burnup. The core flux distribution is monitored on a periodic basis by in-core detector systems designed to measure the neutron flux along the length of fuel assemblies spaced throughout the core. The resulting measurements are capable of producing a three dimensional map of the neutron flux within the core. The flux map is then used to update the burnup and k_{inf} at nodes along every fuel assembly to map the relative fission rates throughout the core. It then calculates the incremental burnup at each node needed to produce the actual heat generated within the core since the last measurement, while also resulting in the most recent flux distribution. Updates of burnup of the fuel assemblies in the core are conducted periodically. The time between updates is made small enough so that changes in the shape of the flux map will not be large and can be easily interpolated. The codes used for these calculations are mature and validated by many reactor years of experience.

An active core load is comprised of two or more batches of fuel that cycle through the core in a shuffled pattern. As an example, at one plant–a large PWR--there are a total of 217 fuel assemblies in the reactor core. All 217 assemblies are removed from the core at each refueling. If Fuel Cycle N-1 is being completed, the next core load would consist of:

- 100 fresh fuel assemblies (Fuel Cycle N), which were recently purchased and accepted.
- 100 once-burned assemblies (Fuel Cycle N-1), which were loaded as new assemblies during the previous refueling outage.
- An additional 17 twice-burned assemblies (from earlier fuel cycles) selected based on their burnup history and their intended position in the core. For example, the center fuel assembly is usually twice-burned.

In preparation for the next fuel cycle, the residual k_{inf} profiles of the 117 partially burned fuel assemblies and the design k_{inf} profiles of the new fuel are used to evaluate the planned core loading pattern against safety limits and to predict the three dimensional flux distribution that should be observed during startup operations.

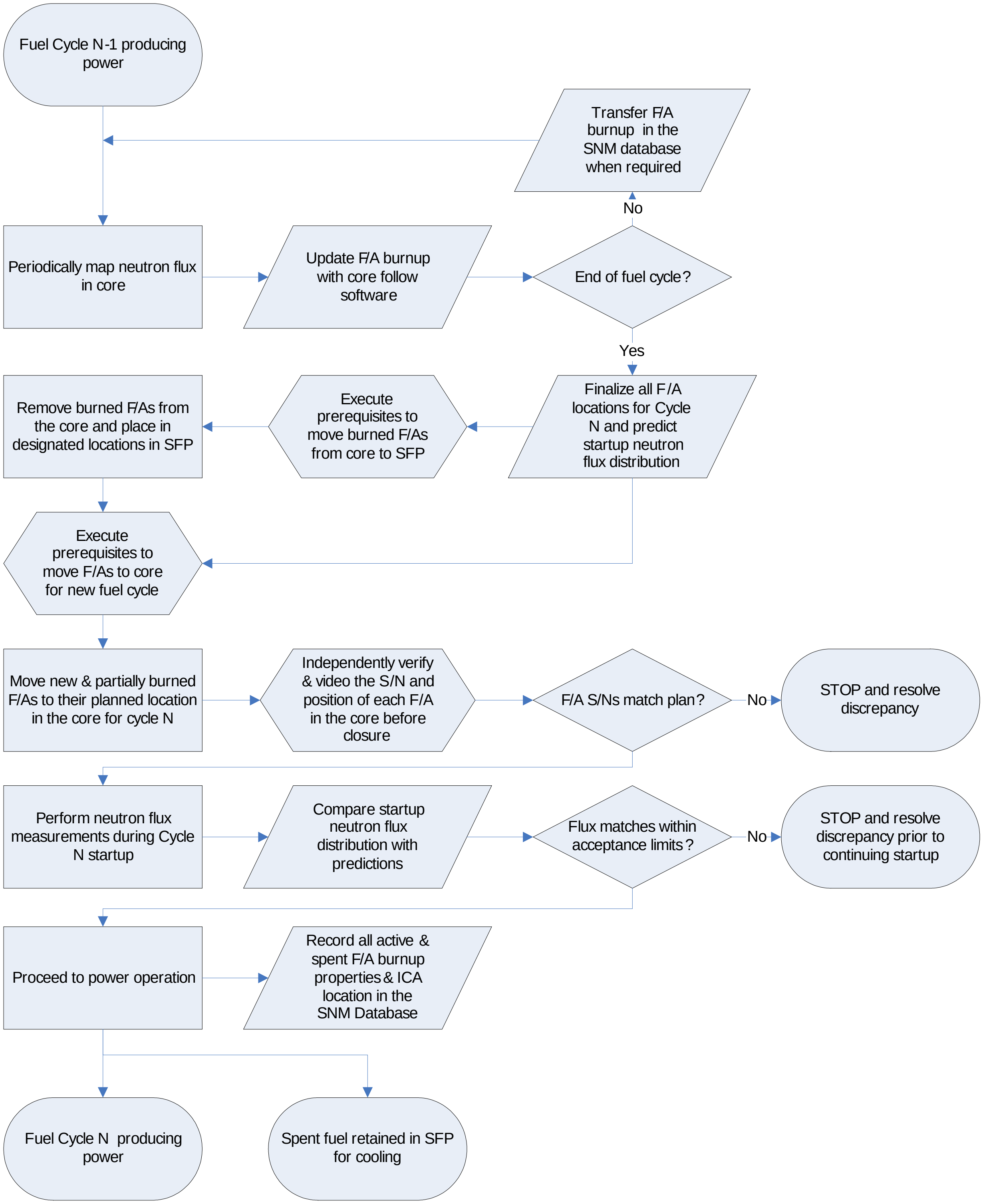

Figure 2-2
Typical Process for Planning and Executing a Core Reload

Fuel transfers within the plant are accomplished using a standard fuel movement procedure, in which a formal fuel transfer document is generated that delineates the serial number, and a

"FROM" position and "TO" position for the transfer. The individual positions in the SFP and core are uniquely defined by a position reference. For example, at one plant, permanent placard strips in the spent fuel and the refueling areas delineate the locations within the SFP and the core. In addition, each SFP and core location is defined against a digital reference.

At one plant, the document is called the Fuel Movement Sequence Data Sheet (FMSDS). During the refueling operation, the movement and placement of fuel assemblies is carefully controlled. The refueling engineer uses three-way communication with the refueling machine operator to identify and verify the "FROM" and "TO" locations of the transfer. An example of three-way communication would be the following exchange:

- First Person: "I identify that the refueling machine is at bridge location AA and trolley location 11."
- Second Person: "I verify that the refueling machine is at bridge location AA and trolley location 11."
- First Person: "That is correct."

Typically, when the core is fully loaded, the refueling engineer and a representative from the Nuclear Oversight and Regulatory Affairs Division use an underwater video camera to conduct an inventory of the core by serial number and location. This inventory is monitored by a qualified person in the control room who verifies the serial number against the FMSDS and core loading map. The refueling engineer and the representative from the Nuclear Oversight and Regulatory Affairs Division must state the correct locations to the control room engineer before the control room engineer responds with the serial number and location to complete a three-way verification. If the refueling engineer or representative from the Nuclear Oversight and Regulatory Affairs Division does not provide the correct serial number and location, the control room will require them to read the serial number and position again until they state it correctly or they agree there is a discrepancy. After loading is complete, a third party reviews the video to independently verify the serial numbers before the upper guide structure is installed.

Following the refueling operations, the approach to criticality provides initial verification of predictions of the k_{eff} of the new core. Any anomaly between the expected responses to the startup sequence will be resolved before proceeding to criticality. Prior to reaching 30% full power, the neutron flux distribution is measured and mapped to verify that the power distribution and peaking factors within the core agree with that predicted by the core-follow software. Discrepancies must be reconciled before the reactor can be authorized to proceed to full power.

In summary, each spent fuel assembly in the SFP has a unique serial number that can be directly associated with its burnup history over multiple fuel cycles. The confirmation of the location of that fuel assembly has been subjected to three-way communication to verify positions during transfer, three-way communication with the control room during verification of serial number and location, and by third-party independent review of the video. Proper agreement of the startup measurements provides confirmation that the residual k_{inf} in the burned fuel assemblies predicted by the core-follow calculations are correct, the new fuel assemblies were manufactured to specifications, and the core was loaded correctly. Finally, the close relationship between new core design and the final burnup update of the previous cycle provides additional confidence that the calculated burnup of the spent fuel assemblies discharged to the SFP are accurate.

2.1.3 Records of Fuel Assembly History

The above discussion indicates that core follow software develops and stores accurate information on a serial number basis of the burnup of spent fuel assemblies discharged to the SFP. A major issue for selecting appropriate fuel assemblies for loading into spent fuel casks is verification that the correct initial enrichment, subsequent burnup history, and time spent in the SFP following its final discharge from the core are accurately recorded by serial number in the SNM accountability software used to select fuel assemblies for loading into a dual-purpose SFC.

10 CFR 74 establishes the requirements for the control and accounting of special nuclear material at fixed sites and for documenting the transfer of special nuclear material. 10 CFR 74.19, *Recordkeeping*, subparagraph (a) requires that

> "(1) Each licensee shall keep records showing the receipt, inventory (including location and unique identity), acquisition, transfer, and disposal of all special nuclear material in its possession regardless of its origin or method of acquisition.
>
> (2) Each record relating to material control or material accounting that is required by the regulations in this chapter or by license condition must be maintained and retained for the period specified by the appropriate regulation or license condition. If a retention period is not otherwise specified by regulation or license condition, the licensee shall retain the record until the Commission terminates the license that authorizes the activity that is subject to the recordkeeping requirement.
>
> (3) Each record of receipt, acquisition, or physical inventory of special nuclear material that must be maintained pursuant to paragraph (a)(1) of this section must be retained as long as the licensee retains possession of the material and for 3 years following transfer or disposal of the material."

Implementation is left to the utility, which may use a combination of procedures, one or more databases, and their supporting software to control all the data elements needed to track burnup and control its F/As.

The January 2007 draft of ANSI 15-8, *Special Nuclear Material Control and Accounting Systems for Nuclear Power Plants*, provides reasonable best practice guidelines for a practical and effective system for tracking SNM within its Item Control Areas (ICAs). *Section 10, Records and Reports* of this document states:

> "Records shall be created and retained. The accounting records are the basis for the material accounting and control program. Quantitative data generated by the utility's calculations of changes in quantities and isotopic composition, due to irradiation and decay, shall be recorded and reported in accordance with the utility's standard recording and reporting procedures. The records and reports system shall include:
>
> > (1) A centralized accounting system;
> > (2) Material control records are maintained for each ICA;
> > (3) Reconciliation of the material control records to the results of physical inventories;
> > (4) Recording the transfer of SNM into, within, or out of each ICA;

(5) Recording the creation of items containing SNM, such as creation of a rod fragment;
(6) Reporting to the central accountability office of the transfer of SNM into, within, or out of an ICA;
(7) Perpetual inventory records of each ICA, including identity and location of each item in the ICA that contains SNM;
(8) Historical data of SNM in each nuclear fuel assembly, fuel component, or non-fuel SNM item while in the possession of the utility; and
(9) Retention for three years following transfer or disposal of the SNM.
(10) For ISFSI ICAs:
(i) retention for five years following transfer or disposal of the SNM
(ii) separate storage locations maintained for two sets of material control records."

The results of this risk assessment illustrate the importance of a centralized accounting system and procedures to support spent fuel handling and storage operations. An interconnected system would enable separate programs to transfer information electronically through defined interfaces that minimize the need for manual clerical tasks subject to human error. In the past, plants have used manually prepared documents to control and account for various movements and processes involving SNM. Since some plants may still use manual methods for transferring data, the risk assessment accounts for the potential for error during this process. However, as the original flux map measurements and burnup calculations are retained as the record of the changes in SNM due to burnup in the reactor core, a comparison of this data with the SNM accountability database can reveal fuel assembly burnup errors before a spent fuel cask is shipped.

The two events described below illustrate the importance of such a system. The corrective actions implemented by the licensees conform to the recommendations of the January 2007 draft of ANSI N15-8. This should greatly reduce the likelihood that misload events due to selection errors will occur in the future.

- Selection of fuel assemblies with insufficient cooling time

  During the 1999 loading of five dry fuel storage casks at a plant, eleven fuel assemblies were loaded which did not meet the Certificate of Compliance (CoC) criteria for post irradiation cooling time greater than or equal to five years, which is specified in Table 1 of the CoC. The fuel assemblies were in compliance with the initial enrichment and burnup limits of the CoC. Although less than five years, the actual post-irradiation time for these eleven bundles was greater than four years at the time they were loaded. At the time the error was detected, all fuel assemblies exceeded the five-year CoC post-irradiation requirement, and the casks were operable. There had been no impact on occupational dose, no impact on the public, and all surveillance parameters were satisfactory.

  The root cause of this incident was weakness in the cask loading spent fuel selection process. The process required the responsible engineer to access multiple documents, including information beyond the formal references, to determine the last irradiation date for each fuel bundle, and there was no working level procedure focused explicitly on the analytical aspects of fuel selection to assist the engineer to ensure the correct data was utilized. The licensee has modified the fuel selection process to include a specific working level procedure for fuel selection. In addition, the reactor engineering fuel management file was upgraded to include fuel cycle date information, and computerized to improve utilization of the information for control of spent fuel operations.

The error was discovered when the utility voluntarily did a review of its spent fuel stored in dry storage casks in response to NRC Information Notice 99-29, which alerted addressees to “pay close attention to the authorized contents for loading into spent fuel casks.” Although no specific action or written report was required, the utility undertook the review.

- Selection of fuel assemblies utilizing incomplete fuel cycle records

While updating a database used to store data associated with assemblies to be loaded into dry casks, it was discovered that several fuel assemblies loaded into a cask 16 months earlier exceeded the decay heat and exposure limits specified in the associated CoC requirements. The loaded fuel assemblies had been selected from a Cask Loader database that contained incomplete fuel assembly burnup data. Corporate fuel engineers compiled the database using fuel assembly burnup data for the first 10 cycles that was supplied by site reactor engineering personnel. Cycle 10 was selected as a cutoff point to ensure that the fuel assemblies had experienced sufficient decay time to meet the cask CoC limits. Once the data was verified by an independent fuel engineer and all associated burnup calculations were completed, the database was used by site reactor engineering personnel assigned to develop the cask loading strategy. However, the database included fuel assemblies that that were used in the reactor after cycle 10. As such, additional irradiation for these assemblies was not accounted for in the database.

At the time the cask was loaded, the licensee did not have a specific procedure to control the development of the database. As result of this event, the licensee developed a standard working level procedure for developing the necessary databases and calculations for selecting and analyzing spent fuel cask loads, along with addressing the scope and method for performing the independent verification. It specifically identifies the information to be included in the database, requirements for updates based on end-of-cycle conditions, and interfaces between corporate and site engineering for transfer of data.

These errors were discovered during an incidental revision of the database by an individual who questioned lower-than-expected burnup values for some of the fuel assemblies.

2.2 Control of Spent Fuel Assemblies Within the Spent Fuel Pool

Figure 2-3 illustrates the life cycle of fuel assemblies after they are discharged from the core into the SFP. Spent fuel assemblies are stored in the SFP for periods exceeding five years to allow the short-lived fission products to undergo radioactive decay before they may be loaded into dry storage spent fuel casks. During this cool down period, annual inventories are conducted in accordance with 10 CFR 74.19(c), which requires that:

> “…each licensee who is authorized to possess special nuclear material, at any one time and site location, in a quantity greater than 350 grams of contained uranium-235, uranium-233, or plutonium, or any combination thereof, shall conduct a physical inventory of all special nuclear material in its possession under license at intervals not to exceed 12 months. The results of these physical inventories need not be reported to the Commission, but the licensee shall retain the records associated with each physical inventory until the Commission terminates the license that authorized the possession of special nuclear material.”

ANSI N15-8 draft guidelines regarding the conduct of the inventory state:

> "7.3 Inventory Method. Count of all items which contain SNM shall be conducted.
>
> 7.3.1 Assemblies and Fuel Component Containers. For fuel assemblies and fuel component containers (as defined above) in the spent fuel pool, an item count is sufficient. If the contents of a fuel component container are accessed, the contents shall be physically re-inventoried before it can be treated as an item for inventory purposes.
>
> 7.3.2 Fuel components. For fuel components that are not part of an intact assembly, physically captured in an assembly, or stored in a fuel component container (such as separated rods stored in baskets, and pellets or fragments stored in a bucket), each component shall be inventoried.
>
> 7.3.3 Sealed Containers. If items are stored in a container that has been with a tamper indicating device, verification of the integrity of the seal is sufficient.
>
> 7.3.4 Reactor. Whenever fuel assemblies are loaded into a reactor, the unique identifier and location of each item shall be visually verified. When the reactor vessel is reassembled, the reactor is considered one item.
>
> 7.3.5 Non-fuel SNM. Where verification is not practical (for reasons such ALARA), an item count shall be taken at the time the item is placed into a designated storage area and administrative procedures shall be established that records concerning the location and unique identity are accurate.
>
> 7.4 Reconciliation. All material control records shall be reconciled to the physical inventory. Discrepancies between physical inventory and the inventory of record shall be investigated and resolved expeditiously. The inventory of record shall be updated to agree with the result of the physical inventory."

The annual inventory verifies that the number of occupied and empty cells in the SFP match that on record in the SNM accountability database. There is no requirement to conduct a visual verification of fuel assembly serial numbers in the SFP during inventories.

During the cool-down period, fuel assemblies may be moved one or more times in order to maintain compliance with the safety limits for the SFP. However, the movement of individual fuel assemblies within the SFP is formally controlled by procedure, and the SNM accountability database is updated to reflect the new locations within the SFP. In addition, the NRC may require a direct verification of fuel assemblies by serial number at any time.

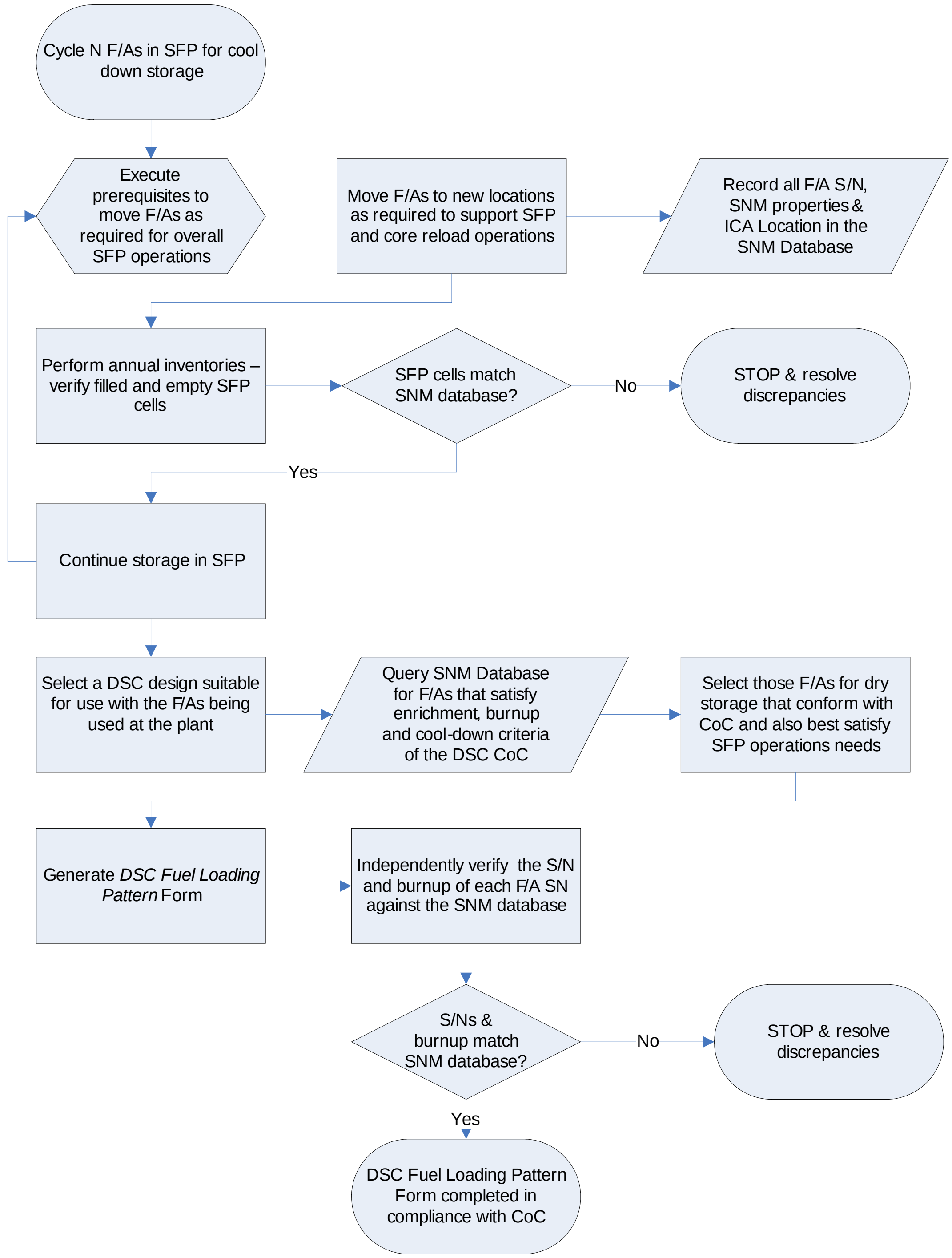

Figure 2-3
Activities during Cool Down of Fuel Assemblies in the Spent Fuel Pool

2.3 Dry Storage of Spent Fuel Assemblies

Independent storage of spent nuclear fuel is regulated by 10 CFR 72, which includes the requirement that a spent fuel cask design have a Certificate of Compliance approved by the NRC. Paragraph 10 CFR 72.236 states the specific requirements for spent fuel storage cask approval and fabrication, and 10 CFR 72.236(m) encourages consideration to be given to compatibility with removal of the stored spent fuel from a reactor site, transportation, and ultimate disposition by the Department of Energy. To satisfy the criticality safety requirements for transportation, the vendor of the cask must demonstrate that the cask will remain subcritical with sufficient margin to compensate for any uncertainties under the accident conditions hypothesized in 10 CFR 71.55, which states the general requirements for fissile material packages. The combination of enrichment and burnup values that meet that requirement constitutes the limits against which all the fuel assemblies to be loaded are compared for criticality safety.

2.3.1 Selection of Fuel Assemblies for Loading into a Spent Fuel Cask

The plant is required to prepare fuel selection documentation for each cask in accordance with the Certificate of Compliance (CoC) issued for the cask. The plant's fuel management group are responsible for assuring that fuel having a given initial enrichment percentage must exceed the burnup limit for that enrichment. As shown in Figure 2-4, the selection process involves accessing SNM accountability database and identifying those fuel assemblies in the SFP that meet the limits associated with initial enrichment, burnup, cool down time, and decay heat load.

The fuel management engineer in coordination with the fuel handling engineer then select fuel assemblies from that list in order to best meet additional criteria based on current and future safety requirements associated with spent fuel operations within both the SFP and ISFSI. As a best case, the software will automatically generate a list of qualifying fuel assemblies in the form of a standard electronic file that contains all the required fuel assembly parameters, based on the entire history of each fuel assembly. Alternately, it may be transferred to a calculation sheet by electronic “cut and paste” or generated manually.

2.3.2 Loading Fuel Assemblies into a Spent Fuel Cask

Once the fuel selection documentation is prepared, it is the refueling engineer's responsibility to load the correct fuel assemblies into the cask. The actions involved in loading a cask are illustrated in Figure 2-4, based on the procedures at an example operating plant. At this plant, DSC is the term used for a spent fuel cask.

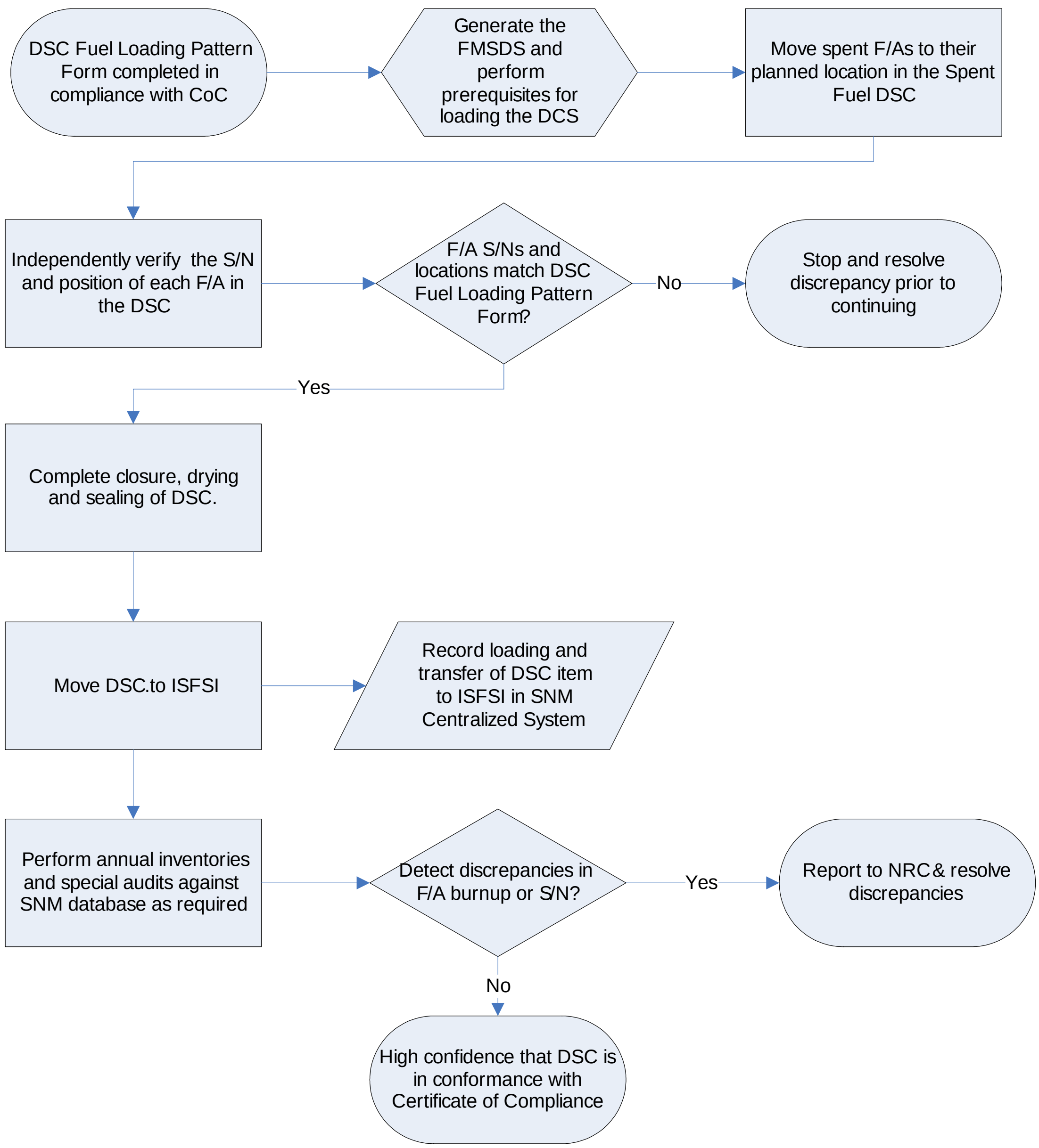

Figure 2-4
Activities Involved in Loading a Dry Shielded (Spent Fuel) Cask with Spent Fuel Assemblies

The first action involves generating a Fuel Movement Sequence Data Sheet (FMSDS) from the DSC Fuel Loading Pattern Form provided by Nuclear Fuel Services. This action is primarily an administrative function involving identifying the location of fuel assemblies having specific serial numbers within the SFP and associating them with the locations within the DSC specified by the DSC Fuel Loading Pattern Form. Locations within the SFP are recorded in plant nuclear fuel inventory software. At the example plant, both the SFP inventory and the layout of the DSC are displayed in graphical format produced by a computer program from the inventory of fuel assemblies in the SFP. The refueling engineer develops the data for the Fuel Movement Sequence Data Sheet by clicking on the storage cell containing the serial number of one of the spent fuel assemblies specified by the DSC Fuel Loading Pattern Form and moving it to its designated location in the layout of the DSC. Once all the fuel assemblies are assigned to a position in the DSC, he prints out the Fuel Movement Sequence Data Sheet, verifies it against the DSC Fuel Loading Pattern Form, and signs the Fuel Movement Sequence Data Sheet.

The Fuel Services Supervisor or his designated representative independently verifies the completed Fuel Movement Sequence Data Sheet against the DSC Fuel Loading Pattern Form. This is accomplished by a visual comparison of the DSC locations of serial numbers specified in the Fuel Movement Sequence Data Sheet with those contained on the original DSC Fuel Loading Pattern Form. There is no specific requirement for the independent verifier to mark each entry on the Fuel Movement Sequence Data Sheet as verified, but the reviewer's signature serves as a record that the review was done correctly.

The refueling engineer or his designee conducts a physical inventory utilizing an underwater video camera to verify that the "FROM" locations in the SFP do, in fact, contain the serial numbers specified in the inventory database. During this inventory, both the location coordinates on the plant's bridge and trolley placards are verified by the two individuals using three-way communication. The serial number is then viewed by the two individuals who confirm that they have read the same serial number using the same three-way communication techniques. At the example plant, the bridge and trolley coordinates are also displayed digitally on the spent fuel handling machine as a double check of the location.

After both the serial number and SFP location of each fuel assembly is verified, the refueling engineer team follows its standard movement procedure to transfer each fuel assembly from its location in the SFP to the designated position in the DSC. As the spent fuel handling machine blocks direct observation of the fuel assembly serial number, the refueling engineer team relies on the location of the spent fuel handling machine to select the correct fuel assembly. The team identifies and verifies both the SFP location and the DSC cell position using the three-way communication protocol previously described. At the example plant, the DSC is loaded from the center out in a crossing pattern that maintains the center of gravity as close to the center of the DSC as possible in order to equalize the canister weight distribution.

Once the DSC is fully loaded, the refueling engineer and the representative from Nuclear Oversight and Regulatory Affairs perform serial number verification by cell location within the DSC using an underwater camera. The camera focuses on the DSC ID number and the drain line location to establish the orientation of the recording. It then follows an identifiable scanning pattern and focuses on each serial number of each fuel assembly long enough for the two individuals to read and verify the serial number against the cell position. The camera scan is video taped.

After completing the verification with the nuclear oversight inspector, the refueling engineer forwards the video recording and the DSC Fuel Loading Pattern Form to an independent third party. This individual reviews the recording to confirm that the DSC fuel loading agrees with the DSC Fuel Loading Pattern. Third-party verification must be completed prior to proceeding with the insertion of the upper spacers and transfer to DSC wash down. The video recording is retained by the plant, enabling an additional review at any time.

2.3.3 Dry Storage in the Independent Spent Fuel Storage Installation

Following closure activities the DSC is transferred to the ISFSI for onsite storage. At any time during the storage period, the records associated with that DSC may be accessed and checked against the SNM accountability database and the core burnup records. In addition, the transfer documents, video records, and annual inventories may be reviewed to verify the serial numbers of the fuel assemblies in each DSC and the SFP. The risk assessment assumes that such a process will be included in the plant procedures. It then assesses the likelihood that such checks would not be done or that a human error of commission during additional checks will result in a failure to detect any errors.

2.4 Assessment of Human Failure Events Associated with Spent Fuel

Figure 2-5 presents the event tree model of the sequence and relationship between the human failure events (HFE) associated with each activity described in Section 2.3. The HFEs are broken into two types identified by the first two letters of the HFE name as follows:

- HA-Errors during human activities that result in a fuel assembly with the wrong serial number being handled during an activity.
- HR-Errors that prevent recovery from previous errors during verification.

Table 2-1 describes the basis and value of each HFE. One HFE is based on historical data while the remainder have been evaluated using the EPRI HRA Calculator software. The details of these HFE evaluations are documented in Appendix A.

Selection errors involve the activities necessary to prepare the DSC Fuel Loading Pattern. The risk assessment hypothesizes an error of commission (HASEL1) involving either a mistake or error of judgment that would result in one or more spent fuel assemblies with reactivity in excess of that authorized by the DSC's Certificate of Compliance being included in the DSC loading configuration documentation. This would result in a misloaded DSC even when all other activities involved in the loading the DSC are conducted correctly. The HEP for HASEL1 corresponds to a nominal human error probability for error of commission when no other human error can be found, as cited in CAL-WHS-MD-000003 REV OOA, Table 2, Item 1, (Reference 12), with a recovery factor to account for independent checking. The resultant HEP of 2.5E-04/cask is below that which might be inferred from the known cask misloads due to selection errors within the 1,000 casks currently in ISFSIs (see Section 2.1.3 of this report). However, the cited misloads involved fuel assemblies with heat generation rates above that permitted by the CoC rather than excess reactivity. In addition, recommendations made in this risk assessment for implementing the recommendations of the January 2007 draft of ANSI 15-8, *Special Nuclear Material Control and Accounting Systems for Nuclear Power Plants*, and establishing written procedures for these processes make the use of the lower HEP for HASEL1 reasonable.

Using the DSC Fuel Loading Pattern, the refueling engineer manually prepares a Fuel Movement Sequence Data Sheet (FMSDS), which is followed by the refueling engineer crew while executing the transfers. While preparing the FMSDS, the refueling engineer can either select a wrong serial number to designate the fuel assembly to be transferred, or place the fuel assembly in a wrong cell within the DSC for any one of the 32 fuel assemblies selected by Nuclear Fuel Management for transfer. The Supervisor, Nuclear Fuel Management, or designee compares the fuel assembly serial numbers assigned to each DSC location on the FMSDS with the DSC Fuel Loading Pattern provided to the refueling engineer (HRFMS1). If an error exists and is not detected, the fuel assembly associated with the wrong serial number will be transferred, and the error will be undetectable until the completed DSC load is again checked against the DSC Fuel Loading Pattern after the DSC is loaded (HRDSC1).

The next HFE, HATRN1 models errors during the actual transfer of the fuel assemblies. As indicated in Table 2-1, it is based on historical evidence that accounts for fuel misload events during either pickup or placement due to selection, procedures, or other errors. Section 6.3.1 of Reference 10 describes the estimation of the HEP based on the Bayesian approach. The calculation uses a non-informative prior distribution updated with evidence of 327 misload errors out of a total of 1,199,000 fuel transfers to arrive at a mean HEP value of 2.73E-04/transfer. Reference 11 provides the description of the evidence for misloading error. It is based on an assessment conducted by Framatome for the period 1985-1999. This initial year was selected because of "inconsistent reporting prior to 1985 and most of the post Three Mile Island (TMI) rules and regulations had been implemented by operating nuclear power plants." The ending year was the most recent complete year of fuel-handling events available when the Framatome study was completed. The fuel-handling events were compiled from a "review [of] all license event reports and other published reference media (e.g., Institute of Nuclear Power Operations (INPO) library database) that pertain to fuel-handling events…" A total of 327 events were categorized misloading events. Attachment I to Reference 11 provides a brief summary of a sample of 91 of the events.

Those events categorized as a misload include a number that may not be applicable to a DSC misload. For example, some fuel assemblies loaded into the core had slightly more uranium than permitted by Technical Specifications because of vendor error. In other cases, the movement of fuel assemblies within the SFP placed them in locations (e.g., next to a wall) not permitted by Technical Specifications. As stated in Assumption 3.1 of Reference 11, the total of 1,199,000 transfers is based on the number of operating cycles at all nuclear power plants, assuming each refueling batch makes up one third of the core. Although the LERs and other sources count errors during all movements, this value for the denominator does not include fuel assembly movements associated with dry cask storage or spent fuel shuffle and re-racks within the SFP, making this total smaller that the total number of movements. Thus, it is judged that both the numerator and denominator provided by the historical evidence contribute to a conservative estimate of misloading error.

The value of the HEP of 8.7E-03/DSC for the HFE HATRN1 reflects the fact that there are 32 opportunities to misidentify the "FROM" location and move an incorrect fuel assembly, so the total likelihood of an error during transfer is 32 times the HEP for a single transfer. Any misloaded fuel assembly is assumed to be placed in the location that would produce the highest k_{eff} in the criticality calculations.

At the example plant, the refueling engineer and spent fuel handling machine operator use three-way communication to recover from an error in positioning the spent fuel handling machine during the transfer of the fuel assembly. However, since this assessment uses an HEP derived from historical data, and the errors occurred despite (unknown) checking procedures in place at the time of the error, it must be assumed that no credit should be taken for recoveries revealed via three-way communication verification.

FMS	Refueling Engineer (RE)	FMS Supervisor	Refueling Engineer and Crew	Refueling Engineer and Rep. from Nuclear Oversight	Third Party	Third Party	
Select F/As for DSC in compliance with CoC	Prepare FMSDS from DSC Fuel Loading Pattern Form	Verify FMSDS S/Ns and DSC locations against DSC Fuel Loading Pattern Form	Individually Transfer 32 F/As from SFP to DSC	Verify F/A S/N against DSC Fuel Loading Pattern Form using 3-Way communication	Independent Review of Video against DSC Fuel Loading Pattern Form	Independent SNM inventories and/or SNM audits prior to shipment	End State
HASEL1	HAFMS1	HRFMS1	HATRN1	HRDSC1	HRDSC2	HRSEL1	

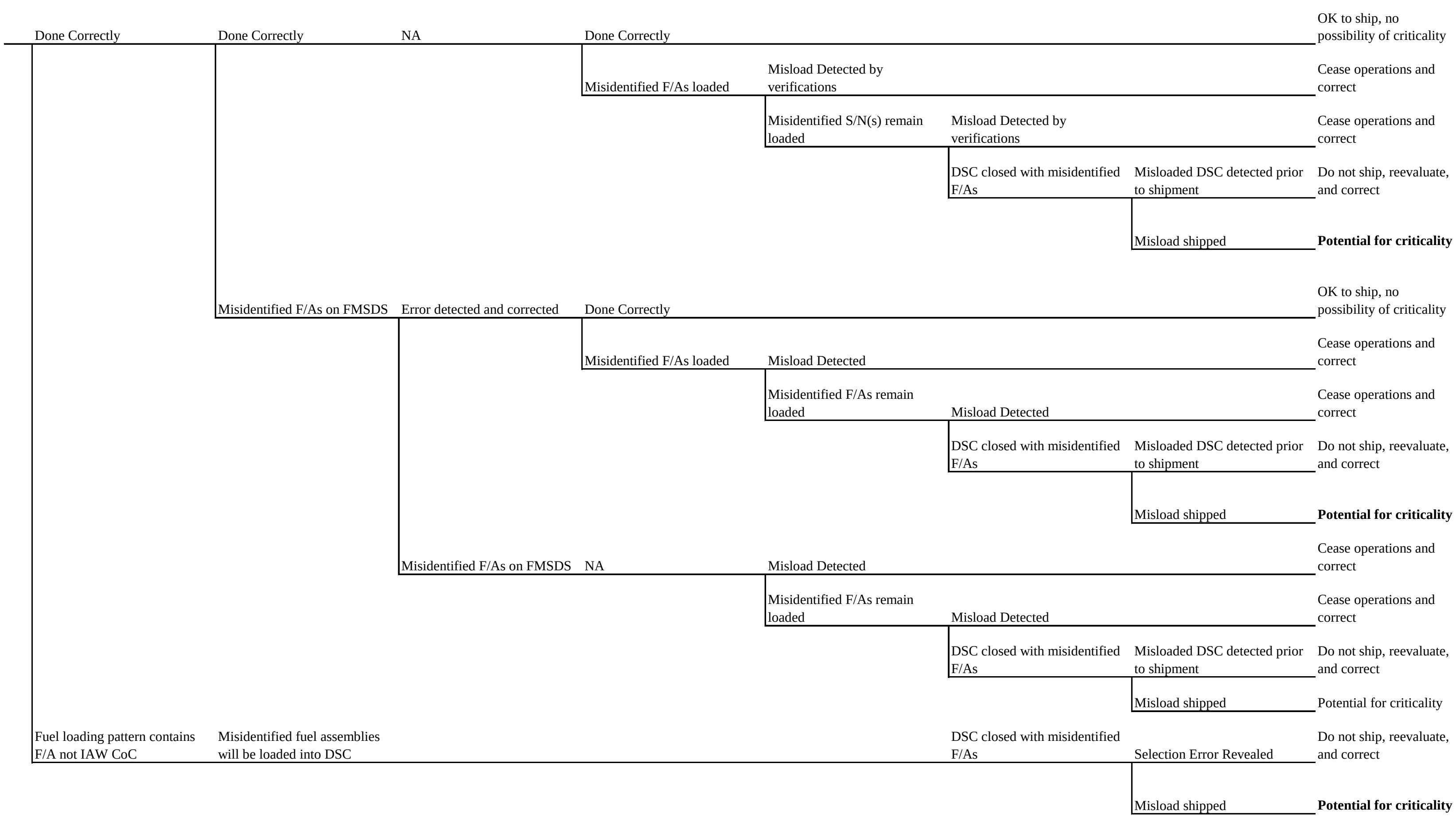

Figure 2-5
Event Tree Logic for the Assessment of Human Failure Events During the Fuel Movement Process

Table 2-1
Summary Description of Human Failure Events during the Loading of a Spent Fuel Cask

HFE Name	Description	Activities Performed	Performance Shaping Factors	HEP	Error Factor
HAFMS1	Prepare Fuel Movement Sequence Data Sheet	This action is primarily an administrative function involving identifying the location of F/As with specific S/Ns within the SFP and associating each with the location within the DSC specified by the DSC Fuel Loading Pattern Form provided by Nuclear Fuel Management. Locations within the SFP are recorded in plant nuclear fuel inventory software as well as on a tag board. The refueling engineer generates the transfer sequence using software designed to assist this process and verifies that the resulting FMSDS correctly delineates the transfer sequence.	This action is done manually in an office environment. At the example plant, both the SFP inventory and the layout of the DSC are displayed in graphical format produced by a computer program. The refueling engineer develops the data for the FMSDS by clicking on the storage cell and moving it to the location in the DSC specified by the DSC Fuel Loading Pattern Form. Although there is no step by step direction for creating the FMSDS, it can be assumed that the refueling engineer is qualified to accomplish the activity without written steps.	1.3E-02	5
HASEL1	Select F/As conforming with CoC	This action is a Nuclear Fuel Management administrative function to generate the DSC Fuel Loading Pattern Form. The initial enrichment and burnup of every F/A selected for loading must be in conformance with the CoC. F/A initial enrichment and burnup recorded in plant SNM accountability software. Burnup data can be verified by reference to the plant's core follow and fuel cycle management software.	This is an administrative action is done in an office environment. The required information can be extracted electronically from the SNM accountability software, but will require manual query and decision activities. Initial HEP is a nominal human error probability for error of commission when no other human error can be found. The action includes recovery from errors by error of commission during active independent verification that the Fuel Loading Pattern Form is correct.	2.5E-04	10

Table 2-1 (continued)
Summary Description of Human Failure Events during the Loading of a Spent Fuel Cask

HFE Name	Description	Activities Performed	Performance Shaping Factors	HEP	Error Factor
HATRN1	Individually Transfer 32 F/As from SFP to DSC	HEP is based on available historical evidence for fuel misload due to selection, procedures, or other errors is 327 out of a total of 1,199,000 fuel assembly movements. [Ref 10 section 6.3.1 based on Ref 11].	Data Based HEP. Refer to the discussion in the text of this section. Overall HEP = 2.73E-04*32 = 8.7E-03 based on 32 opportunities for error.	8.7E-03	3
HRDSC1	Verify "TO" Location of S/Ns on FMSDS in the DSC	Representative from Nuclear Oversight verifies with the refueling engineer via three-way communication the location and serial number of fuel assemblies in the DSC while video taping DSC contents.	Operations are performed in the SFP with no time limits or distractions. Work can be done within one shift, and the qualifications of all operators are verified.	2.8E-03	5
HRDSC2	Independent verification of spent fuel S/Ns in the DSC	A qualified independent person reviews the video recording and verifies the S/N and location of F/As within the DSC against the DSC Fuel Loading Pattern form provided by Nuclear Fuel Management.	The review of the video tape can be done in an office environment. The reviewer observes the S/N and location of each of 32 F/As on the video and compares it to the DSC Fuel Loading Pattern form provided by Nuclear Fuel Services. Although there is no step-by-step direction for accomplishing the transfer, it can be assumed that the reviewer is qualified to accomplish the activity without written steps. As this action is done by one person, the only recovery from an error is self checking.	7.4E-02	5

Table 2-1 (continued)
Summary Description of Human Failure Events during the Loading of a Spent Fuel Cask

HFE Name	Description	Activities Performed	Performance Shaping Factors	HEP	Error Factor
HRDSC3	Independent three-way communication verification of DSC video against DSC Fuel Loading Pattern Form	A qualified independent person views the video and records the S/N and location of each F/A in the DSC on a blank DSC Load Pattern Form. He then verifies the completed form against the original with a second person having the original DSC Load Pattern Form via three-way communication.	The review of the video tape can be done in an office environment. Active recording of the S/Ns and three-way communication provides a mechanism for verifying the S/Ns and locations by active participation of two people.	6.8E-04	10
HRFMS1	Nuclear Fuel Management Verifies S/N and "TO" locations of F/As on FMSDS	This action is primarily an administrative function involving verifying the S/N and location of F/As listed on the FMSDS against those specified by the DSC Fuel Loading Pattern Form provided by Nuclear Fuel Management. The Supervisor, Nuclear Fuel Management, or a designee, compares the FMSDS with the DSC Fuel Loading Pattern Form.	This action is done in an office environment comparing two documents side by side. This is a manual activity requiring careful comparison of 32 individual serial number and locations. However, all of them have a distinct meaning to the reviewer, and they have been subject to calculations to produce fuel selection documentation for the NRC.	6.6E-02	5
HRSEL1	Perform independent SNM inventories and/or audits prior to shipment	The action involves active independent verification of both the serial number and initial enrichment and burnup of all the fuel assemblies in a DSC prior to shipping. The risk assessment has revealed that this would be an effective way to verify the contents of a DSC prior shipment. It could be accomplished at any time during storage in the ISFSI.	This is an administrative action is done in an office environment. The required information is extracted from the SNM accountability software, but will require manual query and comparison activities. The HEP is judged to be equivalent to a basic error of commission during an active checking activity.	1.0E-02	5

The HFE HRDSC1 models errors during the visual verification of the fuel assemblies within the DSC against the original DSC Fuel Loading Pattern provided by Nuclear Fuel Management. The refueling engineer and a representative from Nuclear Oversight and Regulatory Affairs Division visually verify the position and serial number of the fuel assemblies with an underwater video camera. Before the verification can begin, the spent fuel handling machine (SFHM) is secured and the video camera is brought into position, first to identify the DSC and orient the camera, and then to focus on the individual serial numbers of the loaded fuel assemblies. Each person observes the serial number and communicates it to the other using three-way communication. For example,

- First Person: “I identify that the fuel assembly serial number is X2B205.”
- Second Person: “I verify that I identify the fuel assembly serial number as X2B205.”
- First Person: “That is correct.”

Although the refueling engineer participated in the fuel transfer, it is judged that there is no dependence between the misidentification errors committed during the visual inspection and previous errors. This is based on the presence of a verifier from another organization and the time required to set up and to initiate the video camera.

The independent review of the video tape of the DSC scan involves comparing the fuel assembly serial numbers and locations of the completed load pattern displayed on the video tape record with the original DSC Fuel Loading Pattern provided by Nuclear Fuel Management. If the independent review is accomplished by one person, the following scenario is hypothesized and modeled by HFE HRDSC2. The independent reviewer is provided the original DSC Fuel Loading Pattern, which indicates the serial number of each fuel assembly at the location in which it resides. The reviewer then observes the video and observes each fuel assembly location and serial number. He then checks each location and serial number against the DSC Fuel Loading Pattern and signs the verification form once the review is complete. The reviewer will most likely check his work, but errors of self-checking are subject to high dependency with the previous errors.

As discussed in Section 2.1.3, 10 CFR 74 requires that each licensee keep records showing the receipt, inventory (including location and unique identity), acquisition, transfer, and disposal of all special nuclear material in its possession regardless of its origin or method of acquisition be retained as long as the licensee retains possession of the material and for three years following transfer or disposal of the material. Therefore, selection and loading errors can be detected by verification of the loaded configuration directly against the fuel assembly burnup records centralized SNM control system prior to shipment. The risk assessment accounts for this ability by including an action to verify fuel assembly burnup records against the fuel assembly serial numbers in a DSC prior to its shipment offsite. It is recommended that it be done either as part of an annual inventory or just prior to shipment; the risk assessment includes the HFE for failing to accomplish such an audit as HRSEL1.

2.5 Results and Observations

Figure 2-6 presents results of the quantification of the event tree. It can be seen on the event tree that there are three opportunities to create a misloaded DSC.

- An error of commission during the selection of fuel assemblies for loading into the DSC produces a loading pattern of fuel assemblies that violates the CoC with respect to initial enrichment and burnup.
- The refueling engineer inadvertently enters an incorrect fuel assembly serial number while preparing the FMSDS from the DSC Fuel Loading Pattern and does not detect it during a self check.
- A misidentification error is made during the transfer of the fuel assemblies from their storage position in the SFP to the DSC, leading to an incorrect fuel assembly being loaded. As stated earlier, this error rate is based on historical evidence that included a wide variety of events, most of which involved errors that would not impact the criticality of a spent fuel cask.

The event tree shows the importance of thorough active independent verification of the fuel selection process as well as the actual cask loading. Without this verification the likelihood of shipping a misloaded SFC would be 2.5E-04/cask. 10 CFR 74.19(a)(3) requires that each licensee maintain a complete record of fuel burnup and movement as long as the licensee retains possession of the material and for three years following transfer or disposal of the material. Therefore, such a verification can be implemented at any time up to the shipment of the spent fuel off site.

A formal process, similar to that used during the actual transfer of fuel assemblies, that requires independent verification to actively record the data and use of three-way communication techniques provides the most effective way to detect any errors during the selection and loading process. This method avoids any tendency to “see what one thinks is correct” when making a passive comparison of data. To illustrate the advantage of such a technique, the human failure event HRDSC3 was hypothesized to account for the additional activities. The independent reviewer is provided a blank DSC Fuel Loading Pattern template, such as the one shown in Figure 2-7. The reviewer then observes the video and records the serial number in the appropriate DSC location on the blank template. Once the scan is completed, the reviewer reads the serial number and location of each fuel assembly to a second person who has the original DSC Fuel Loading Pattern, who verifies that it is correct. Once agreement is verified, both people sign the verification. When compared to HRDSC2, the requirement for separate people to view the verification and original documents and agree that they are the same reduces the error rate by a factor of about 100. This technique should be considered for all verification actions.

FMS	Refueling Engineer (RE)	FMS Supervisor	Refueling Engineer and Crew	Refueling Engineer and Rep. from Nuclear Oversight	Third Party	Third Party	
Select F/As for DSC in compliance with CoC	Prepare FMSDS from DSC Fuel Loading Pattern Form	Verify FMSDS S/Ns and DSC locations against DSC Fuel Loading Pattern Form	Individually Transfer 32 F/As from SFP to DSC	Verify F/A S/N against DSC Fuel Loading Pattern Form using 3-Way communication	Independent Verification via Review of Video against DSC Fuel Loading Pattern Form	Perform independent SNM inventories and/or audits prior to shipment	Scenario Likelihood
HASEL1	HAFMS1	HRFMS1	HATRN1	HRDSC1	HRDSC2	HRSEL1	
9.998E-01	9.9E-01	NA	9.9E-01	NA	NA	NA	OK
			8.7E-03	9.97E-01	NA	NA	OK
				2.8E-03	9.3E-01	NA	OK
					7.4E-02	9.9E-01	OK
						1.04E-02	1.9E-08
	1.3E-02	9.3E-01	9.9E-01	NA	NA	NA	OK
			8.7E-03	9.97E-01	NA	NA	OK
				2.8E-03	9.3E-01	NA	OK
					7.4E-02	9.9E-01	OK
						1.04E-02	2.3E-10
		6.6E-02		9.97E-01	NA	NA	OK
				2.8E-03	9.3E-01	NA	OK
					7.4E-02	9.9E-01	OK
						1.04E-02	1.8E-09
2.50E-04						9.9E-01	OK
						1.04E-02	2.6E-06

Total likelihood of a spent fuel cask shipment with one or more misloaded F/As = 2.6E-06

Figure 2-6
Quantification of Human Failure Events Leading to a Misloaded Dry Spent Fuel Cask

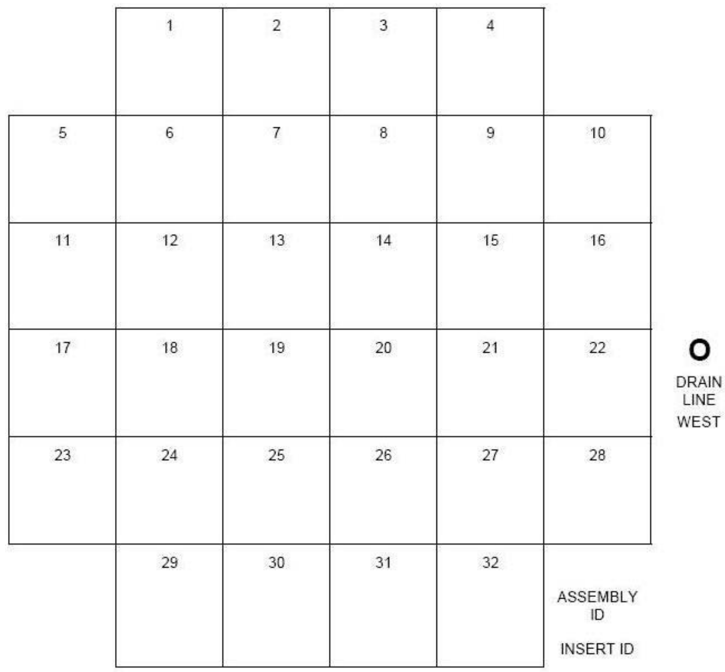

Figure 2-7
Example of a Template for Recording Fuel Assembly Serial Numbers during the Independent Verification Process

The likelihood of a misloaded DSC can be reduced further by eliminating manual preparation of the DSC Fuel Loading Pattern and the FMSDS. The database structure of fuel inventory programs permits the transfer of data between programs that can generate prospective cask loading patterns, support the fuel management and safety requirements for the SPF, and perform the analyses necessary to meet cask criticality and heat loads safety requirements. If the plant predefines its criteria for the DSC loading sequence, the FMSDS could be generated as well as the DSC Fuel Loading Pattern. This would essentially eliminate the Top Event HAFMS1 from the event tree.

3
FREQUENCY OF TRAIN ACCIDENTS

3.1 Overview of Spent Nuclear Fuel Cask Shipping via Railroad

Spent nuclear fuel shipping casks have been used to transport spent nuclear fuel used in nuclear power plants and research reactors to interim storage sites. Each shipping container is designed to maintain its integrity under normal and hypothetical accident conditions.

In the United States, the acceptability of the design of each cask is judged against Title 10, Part 71, of the Code of Federal Regulations. The designs must demonstrate (e.g., via computer modeling and/or testing) protection against radiological release to the environment under all of the following hypothetical accident conditions, which generally encompass >99% of all accidents:

- A 9-meter (30-foot) free fall on to an unyielding surface
- A puncture test allowing the container to free-fall 1 meter (about 40 inches) onto a steel rod 15 centimeters (about 6 inches) in diameter
- A 30-minute, all-engulfing fire at 800 degrees Celsius (1475 degrees Fahrenheit)
- An 8-hour immersion under 0.9 meters (3 feet) of water
- Further, an undamaged package must be subjected to a one-hour immersion in 200 meters (655 feet) of water

Between 1975 and 1977 Sandia National Laboratories conducted full-scale crash tests on spent nuclear fuel shipping casks. Although the casks were damaged during the tests, experts determined that none would have leaked.

The U.S. Department of Transportation (DOT) has the primary responsibility for regulating the safe transport of radioactive materials in the United States. The DOT and NRC specify, via 10 CFR Part 71, 49 CFR Parts 107, 171-180, and 390-397, and via the NRC nuclear materials transportation shipping requirements web page, that licensees and carriers involved in spent fuel shipments comply with a number of rigorous requirements, including the following:

- A shipper must use NRC-approved routes for transport of spent nuclear fuel.
- The shipper must make sure that spent fuel is protected against radiological sabotage.
- Shippers who transport or deliver spent fuel to a carrier for transport are required to meet specific requirements that include
  - notifying the NRC of the shipment,

  – having procedures for addressing emergencies,
  – having a communications center,
  – having a written log of shipment events,
  – making arrangements with local law enforcement agencies for shipments while en route, and
  – using armed escorts in heavily populated areas.

- The time and date of the shipment must be protected as sensitive safeguards information to guard against any act that could threaten the shipment.

Approximately 3,000 shipments of spent nuclear fuel have been transported safely over the U.S.'s highways, waterways, and railroads. Figure 3-1 shows a typical spent nuclear fuel shipping cask mounted on a railroad car.

Figure 3-1
Typical Spent Nuclear Fuel Shipping Cask Mounted on a Railroad Car

3.2 Identification of Accidents Capable of Producing a Criticality Event

NRC-sponsored studies to assess the frequency of radioactive material release during shipments of spent nuclear fuel have estimated the frequency of train accidents by analyzing U.S. Department of Transportation (DOT) Federal Railroad Administration (FRA) accident report data. Two references contain most of the required information, NUREG/CR-4829 (Reference 1), known as the “Modal Study,” and NUREG/CR-6672 (Reference 2).

The Modal Study used historical data to estimate both the frequency of train accidents per train-mile and the fractions of those accidents that result in various damage states. The project team for this EPRI-sponsored project has analyzed historical information using current data available from the FRA. The damage configurations following an accident have been evaluated quantitatively utilizing the key event sequences from the railway accident event tree documented on page 7-12 of NUREG/CR-6672, reproduced as Figure 3-2, to estimate the likelihood of accidents that could lead to the introduction of water into the transportation cask.

Review of the Modal Study, NUREG/CR-6672, and other related documents like NUREG-0170 and NUREG/BR-0111, indicates that event Sequences 3 and 8 from the railway accident event tree applied in those studies (see Page 3-29 of NUREG/CR-4829, Page 7-12 of NUREG/CR-6672, and Figure 3-2) are the key sequences leading to accidents of interest (AOIs) for producing a criticality event.

- Event Sequence 3 is a train accident scenario that results in a train derailment occurring over a bridge with transported cargo being dropped into a body of water (e.g., a river, lake, etc.).
- Event Sequence 8 is a train accident scenario that results in a train derailment on or over an embankment with transported cargo being dropped into a drainage ditch.

3.3 Logic Model

Although the Modal Study provides much information on the models used to assess damage to transportation casks, it does not provide the details supporting the estimate that "the probability of a rail cask's having a structural response greater that 2% strain (S_2) and becoming submerged in water is estimated to be 0.00000078% [7.8E-09], given an accident." (The full quote is provided in Section 1.2 of this report.) Based on this quote, the criticality risk assessment in Section 1.4 assumed that the term *given an accident* encompassed all accidents involving all trains at all speeds. Therefore, the factor of 7.8E-09 should include the values of all of the events in the accident event tree shown in Figure 3-2 plus the effect of speed on the probability of damage and the likelihood that a sufficient depth of water will be present at the final resting place to allow sufficient moderation and reflection to achieve criticality.

This section develops a logic model to examine a variety of cases to shed light on the frequency of accidents that would be more closely associated with the transportation of spent fuel casks, which forms the basis for evaluating the likelihood of damage and inleakage necessary for criticality.

Figure 3-3 illustrates a slightly expanded version of the railway accident event tree logic for Sequences 3 and 8 shown in Figure 3-2. It breaks down the sequence of events to include both the conditional probability that the SNF cask will enter water as a result of an accident and become fully immersed. For purposes of this study, we have assumed that the term "drainage ditches" in Sequence 8 also encompasses all potential adjacent bodies of water (e.g., rivers), which may or may not contain water at the time of the accident. This is an important assumption, as Figure 3-2 does not explicitly address accidents that occur adjacent to bodies of water in which a cask could impact a hard surface, become damaged, and careen into the water.

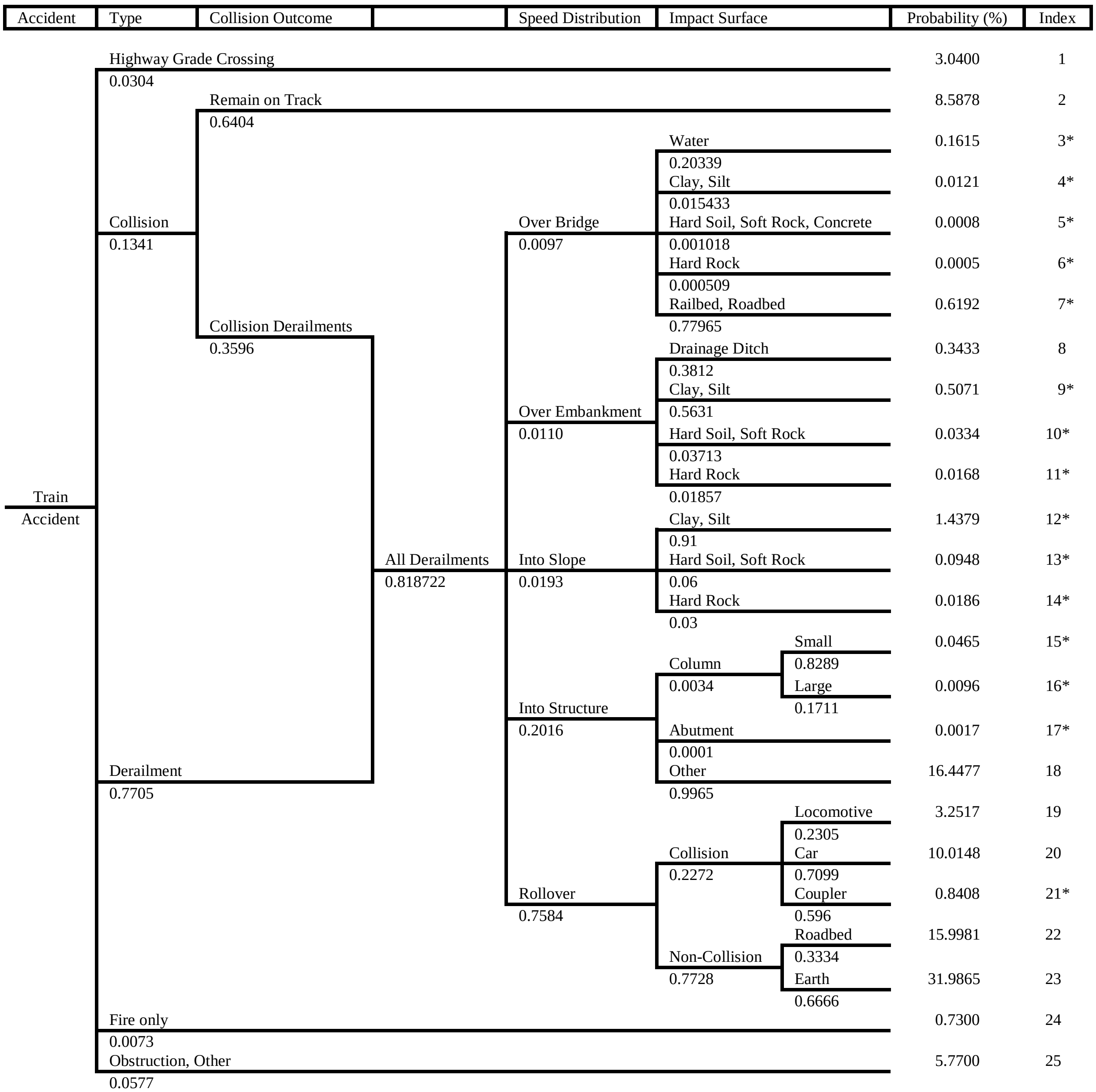

Figure 3-2
Modified Modal Study Train Accident Event Tree (from Page 7-12 of Reference 2)

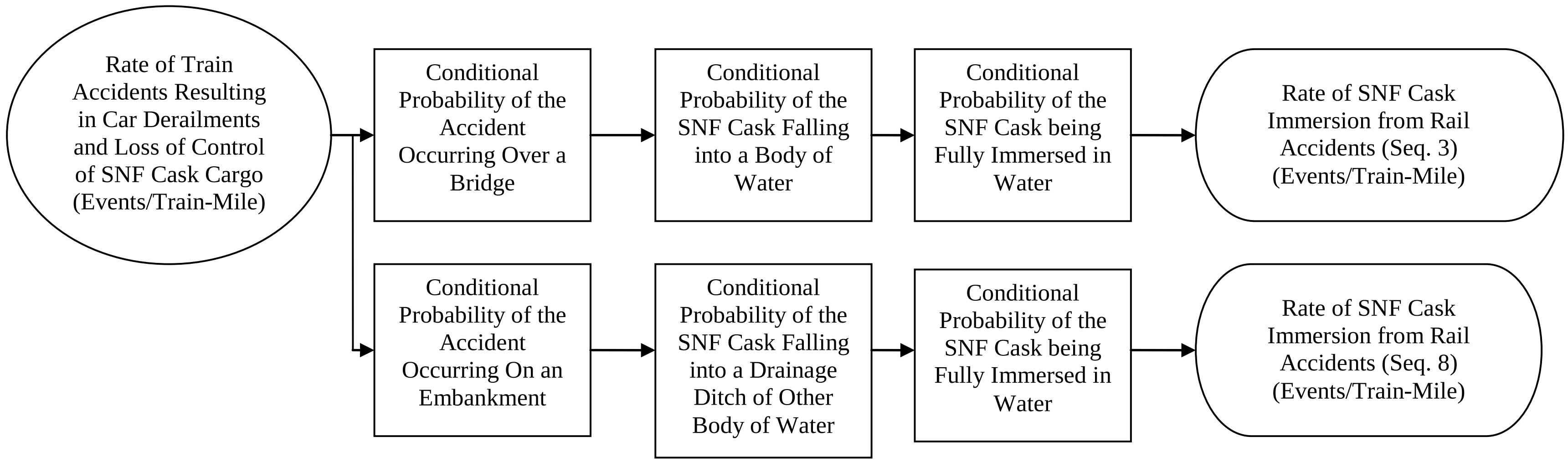

Figure 3-3
Modified Event Sequence 3 and 8 Logic

3.4 General FRA Data Analysis

The FRA requires that all railroads report accidents or incidents resulting in property loss greater than a prescribed dollar threshold via Form F6180.54. Collisions, derailments, fires, explosions, acts of God, or other events involving the operation of railroad on-track equipment (standing or moving) and causing reportable damage greater than the reporting threshold for the year in which the accident/incident occurred must be reported using Form FRA F 6180.54. The property loss threshold for required reporting is reviewed annually and revised, as necessary, by the FRA in accordance with 49 CFR Part 225. The reporting threshold for calendar year 1975 was $1,750.00, and for calendar year 2003 (the most recent year this threshold is reported via FRA web site information) was $6,700.

The FRA database can be accessed at http://safetydata.fra.dot.gov/OfficeofSafety/. In its current form, it is capable of providing information applicable to the assessment of spent-fuel transportation risk, specifically 1) accident rates for only freight trains, 2) the speed at which freight train accidents have occurred, and 3) the fraction of freight train accidents in which HAZMAT cars were present and damaged.

The first portion of the data analysis involved downloading appropriate FRA data and reviewing them for applicability to this PRA. To this end, the project team downloaded all FRA data from the FRA Rail Equipment Accidents Data (Form F6180.54) tables and all FRA Operational Data (Form F6180.55) tables, for all available time periods (January 1, 1975 through May 31, 2006), into a relational database, which was then used to perform all special project data queries for this PRA. The project team also downloaded and reviewed the FRA Data Reporting Guide and the FRA data table file structure and input field specifications for Forms F6180.54 and F6180.55. The file structure and input field specifications for Forms F6180.54 and F6180.55 are presented in this report as Appendices B and C, respectively. These two appendices provide excellent descriptions of the data fields available for data querying applications in this study.

The project team reviewed general accident data and operational (train-mileage) data for the January 1, 1975 to May 31, 2006 time period to determine general trending for railroad accident rates over time. Table 3-1 shows the total count data for all train accidents and train-miles traveled during this time period.

Table 3-1
General Train Accident and Operational Data (1975-May 2006)

Year	Number of All Reported Accidents	Total Train-Miles	General Train Accident Rate (Events/Train-Mile)
1975	8,041	755,244,439	1.06E-05
1976	10,248	774,650,922	1.32E-05
1977	10,362	750,042,291	1.38E-05
1978	11,277	751,964,275	1.50E-05
1979	9,740	763,428,674	1.28E-05
1980	8,451	717,661,741	1.18E-05
1981	5,781	676,216,511	8.55E-06
1982	4,589	573,368,609	8.00E-06
1983	3,906	558,190,305	7.00E-06
1984	3,900	592,600,037	6.58E-06
1985	3,430	570,910,626	6.01E-06
1986	2,761	567,098,523	4.87E-06
1987	2,647	581,313,555	4.55E-06
1988	3,051	609,334,435	5.01E-06
1989	3,080	620,598,940	4.96E-06
1990	3,045	608,837,284	5.00E-06
1991	2,814	576,834,890	4.88E-06
1992	2,531	593,703,777	4.26E-06
1993	2,785	613,973,971	4.54E-06
1994	2,669	655,083,056	4.07E-06
1995	2,619	669,823,264	3.91E-06
1996	2,584	670,923,960	3.85E-06
1997	2,560	676,716,407	3.78E-06
1998	2,745	682,894,841	4.02E-06
1999	2,924	712,452,725	4.10E-06
2000	3,193	722,876,632	4.42E-06
2001	3,240	711,549,906	4.55E-06
2002	2,944	728,674,146	4.04E-06
2003	3,185	743,524,791	4.28E-06
2004	3,591	770,712,759	4.66E-06
2005	3,408	790,669,968	4.31E-06
2006 (through May)	1,218	335,027,877	3.64E-06
TOTAL	139,319	21,126,904,137	6.59E-06
TOTAL (2000 – May 2006	20,779	4,803,036,079	4.33E-06

Figure 3-4 shows the general accident per train-mile rate trend for all reported train accidents and train-miles reported in the FRA databases.

Figure 3-4
Railroad Total Accident Rate Trend (1975-May 2006)

Figure 3-4 shows a significant decrease in the general train accident rate over time, from a peak of 1.50E-05 events/train-mile in 1978 to the current (May 2006) rate of 3.64E-06 events/ train-mile (a reduction of approximately 76%). The current rate appears to have stabilized at approximately 4.00E-06 events/train-mile.

3.4.1 Comparison of FRA Database with the Reference Studies

Table 3-2 compares the overall accident rates reported in the two NRC-sponsored reports with the accident rates obtained directly from the FRA database. It can be seen in Table 3-2 that the total accident rates obtained from current FRA database match those in the original report. This provides high confidence that the current study is using data consistent with the previous studies.

Table 3-2
Comparison of Train Accident Rates Applied in SNF Cask Shipping Accident Studies

			Extracted from FRA On-Line Database		
Source Document	**Data Time Period**	**Reported Accident Rate (Events/Train-Mile**	**Total Accident Rate (Events/Train-Mile)**	**Number of Reportable Accidents**	**Total Railroad Train-Miles (Passenger and Freight)**
Modal Study (NUREG/CR-4829)	1975-1982	1.19E-05	1.19E-05	68,489	5,762,577,462
Reexamination of Spent Fuel … (NUREG/CR-6672)	1988-1995	4.57E-06	4.57E-06	22,594	4,948,189,617
This Report (EPRI)	2000-May 2006		4.33E-06	20,779	4,803,036,079

The project team made the decision to apply January 1, 2000 through May 31, 2006 FRA data to serve as the basis for train accident rate prediction in this PRA. The following factors were considered in making this decision:

- A number of major railway safety measures have been implemented in the railroad industry since the 1970s, and railway safety has improved significantly since then, as shown in Figure 3-4.
- FRA database structures, associated FRA data reporting guidelines, and the general quality of data reporting practices and supporting software have improved over time since the original Modal Study was published.
- It is useful for this analysis that freight train data be separated out from all train data. Freight train operational (train-mileage) data have been reported in such a way that they can be effectively separated from other train operational data for data analysis purposes only since January 1997.

3.4.2 Freight Train Accident Rates

SNF casks will be shipped via trains with design and operational characteristics more closely associated with general freight trains than with other types of trains included in the FRA database, such as passenger trains and railroad yard working trains. Therefore, the project team determined that freight train accidents and freight train operational (train-mileage) data would more realistically estimate accident rates associated with SNF shipments. Table 3-3 shows the count data for freight train accidents and freight train-miles traveled during the 2000-May 2006 time period.

Table 3-3
Freight Train Accident and Operational Data (2000–May 2006)

Year	Number of Freight Train Accidents	Total Freight Train-Miles
2000	1,601	549,382,910
2001	1,599	537,801,654
2002	1,416	546,885,449
2003	1,446	561,202,409
2004	1,593	583,942,080
2005	1,490	597,940,859
2006 (through May)	552	258,290,583
TOTAL	9,697	3,635,445,944

This results in an average freight train accident rate of 2.67E-06 events/train-mile, which is significantly lower than that for all train types of 4.33E-06 events/train-mile.

One of the fundamental assumptions of the Modal Study and NUREG/CR-6672 was that a rail car derailment would have to occur in order for a rail cask to be submerged in water. A primary derailment is one that was encoded as the accident type in the FRA database. A secondary derailment is one in which derailed cars are recorded for accidents encoded with accident types other than derailments (e.g., collisions or other impacts). Applying this criterion to 2000–May 2006 freight train accident data, a total of 8,175 accidents were encoded as primary or secondary derailment events in the FRA database, as shown in Table 3-4. The resulting frequency of derailment accidents is 2.25E-06 per freight train-mile.

Table 3-4
Freight Train Primary and Secondary Derailment Accidents

Year	Primary Derailment Accidents	Secondary Derailment Accidents	Total Derailment Accidents	Total Freight Train-Miles
2000	1,258	109	1,367	549,382,910
2001	1,245	99	1,344	537,801,654
2002	1,103	95	1,198	546,885,449
2003	1,123	99	1,222	561,202,409
2004	1,249	95	1,344	583,942,080
2005	1,146	92	1,238	597,940,859
2006 (through May)	431	31	462	258,290,583
TOTAL	7,555	620	8,175	3,635,445,944

Next, as speed is required to produce the kinetic energy necessary for cask damage, the project team evaluated the accident rate by speed reported in the FRA database. Companies reporting train accidents to the FRA are required to list the last known speed of their trains at the time of an accident, and this information is included in a field (named TRNSPD) of the FRA accident database. There is also a field in the database for the track class (named TRKCLAS) on which the accident occurred. In order to properly correlate the number of accidents at various speeds with the exposure of trains traveling at those speeds, the team estimated train speed by breaking down the total freight train-miles by track class. Freight train speed limits vary by track class, as presented in Table 3-5.

Table 3-5
Train Speed Limits by Track Class in the United States

	Maximum Speed (MPH)	
Track Class	**Freight Trains**	**Passenger Trains**
X	10	Prohibited
1	10	15
2	25	30
3	40	60
4	60	80
5	80	90
6	110	110
7	125	125
8	160	160
9	200	200

The FRA database does not report operational train-mile data by track class. However, Table 2-4 of *Guidelines for Chemical Transportation Risk Analysis* (Reference 8), replicated in Table 3-6, presents an estimated breakdown of train-miles by track class. This data was applied in this analysis to estimate train-miles traveled by track class during the 2000–May 2006 time period.

Table 3-6
Percentage Breakdown of Train-Miles Traveled by Track Class

Track Class	Millions of Main Line Train Miles/Year (1985-1988)	Percent of Total
Class 2	38	9%
Class 3	102	25%
Class 4+	270	65%
Other	4	1%
Total	414	100%
Source: Table 2-4 of *Guidelines for Chemical Transportation Risk Analysis*. [Reference 8].		

NUREG/CR-6672 (p. 7-19) states that finite element analyses indicate that the effective minimum speed a SNF cask would have to be traveling in order to cause seal leakage or breaching of the seal upon impact into an unyielding surface is approximately 60 MPH. Therefore, this would indicate that accident scenarios in which relative velocities of less than 60 MPH between SNF casks and impact surfaces are experienced generally do not result in cask seal leakage or cask failure. Thus, no inleakage of water would occur if the casks become immersed in water. Figure 3-5 presents a chart of freight train derailment accidents by track class and speed range. This figure shows that derailments occurring at speeds over 60 MPH have only occurred on class 4 or greater (4+) tracks. This result is reasonable, as in order for the train to have an accident on class 3 or lower track, it would have to be exceeding the maximum speed limit by at least 20 mph.

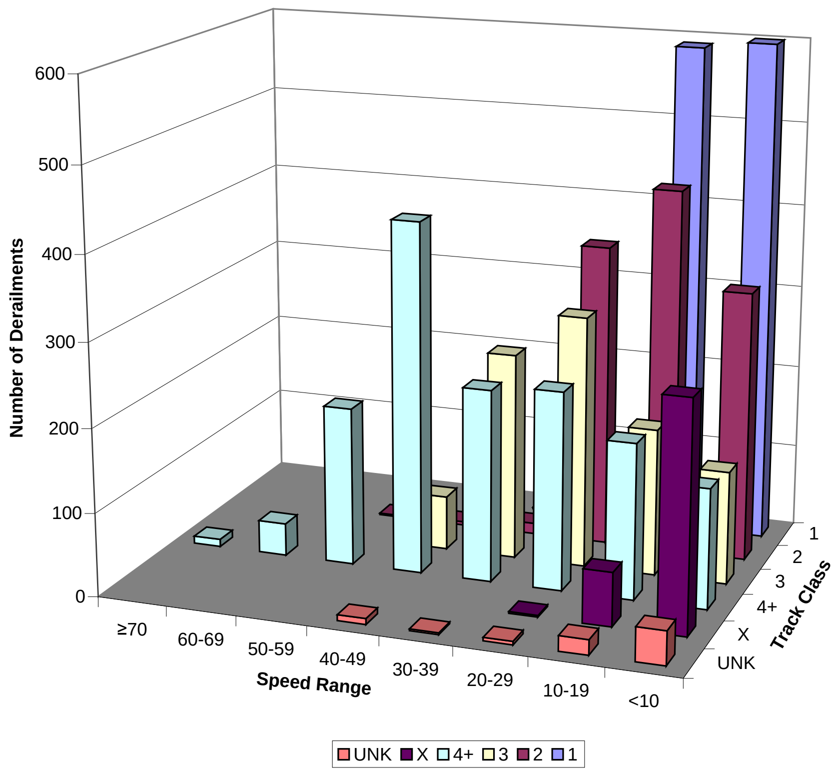

Figure 3-5
Freight Train Accidents by Speed Range and Track Class (2000-May 2006)

It is conceivable that, in some limited number of accidents, derailments could result from collisions of two trains where the relative speed of impact is greater than 60 MPH, but the speed of each train is less. To assess the impact of this possibility, the project team calculated the frequency of freight train accidents based on train-miles accumulated on class 3 and higher track, which permit a speed greater than 30 MPH. This frequency was calculated to be 6.51E-07 events per freight train-mile based on 2000-May 2006 FRA data. Note that the freight train speed limits for track classes X, 1, and 2 are all less than 30 MPH. However, Figure 3-5 indicates that some accidents occurred with indicated speeds above the associated track class speed limit. For purposes of this study, we have counted all accidents that occur at speeds judged capable of producing damage to the SNF casks, but we have limited the train-mile exposure to just those track classes at which that speed is authorized. This approach makes our estimates conservative.

The Modal Study and NUREG/CR-6672 [References 1 and 2] analyzed distributions of train speed for all reported accidents in the FRA data. Table 3-8 and Figure 3-6 show a similar analysis performed in this study using 2000–May 2006 FRA data. For comparison purposes, they also include the corresponding data from Table 5.2 of the Modal Study. Figure 3-6 presents the cumulative distribution for all train collision, derailment, and other impact accidents occurring at train speed ranges consistent with those applied in References 1 and 2. It clearly shows that the over 98% occur at speeds below 60 MPH, and therefore, may not be able to generate enough SNF cask velocity to fail the cask containment function upon impact (i.e., create > 2% strain on the cask). Also, the percentage of train accidents occurring at relatively low speeds is even greater than that found in the Modal Study.

Table 5.2 of NUREG/CR-4829 states that downward velocities due to gravity and bridge height do not exceed approximately 57 MPH, based on postulated cask drops from the largest credible bridge heights located on approved SNF cask rail shipment routes. This indicates that, in order for a dropped cask to experience enough strain to fail its containment function, the train speed at the time of the accident must be a significant velocity vector contributor to the force required to result in at least 2% strain on the cask. Therefore, indicated train speed at the time of the accident should provide a good criterion for analyzing the probability of cask containment function failure as the result of train accidents over bridges or on embankments and is applied as a criterion for some of the case studies performed in Section 3.5.

Table 3-7
Cumulative Probability of All Train Accidents Occurring at or Below Various Speeds

FRA Database 2000 – May 2006				Modal Study	
Train Speed (MPH)	**Collision Accidents**	**Primary Derailment Accidents**	**All Derailment Accidents**	**Collision Accidents**	**Derailment Accidents**
0	0	0	0	0	0
2	0.28209	0.11796	0.15391	0.09385	0.07543
6	0.70451	0.50953	0.52817	0.26286	0.22036
10	0.86812	0.78243	0.75781	0.40788	0.35480
14	0.88858	0.80902	0.78148	0.53042	0.47634
18	0.91326	0.83424	0.80617	0.63240	0.58341
22	0.93018	0.86083	0.83019	0.71598	0.67534
26	0.94288	0.89508	0.86031	0.78345	0.75225
30	0.95557	0.91594	0.88175	0.83709	0.81495
34	0.96262	0.92554	0.8913	0.87908	0.86477
38	0.96968	0.94116	0.90903	0.91147	0.90385
42	0.97673	0.95829	0.92965	0.93606	0.93246
46	0.98166	0.97105	0.9457	0.95446	0.95386
50	0.99013	0.9876	0.96922	0.96801	0.96920
54	0.99224	0.99154	0.97455	0.97784	0.97991
58	0.99295	0.9952	0.98049	0.98486	0.98720
62	0.99788	0.99728	0.98735	0.98980	0.99204
66	0.99788	0.99792	0.98979	0.99323	0.99516
70	0.99788	0.99971	0.99406	0.99557	0.99713
74	0.99788	0.99971	0.99451	0.99714	0.99834
78	0.99788	0.99986	0.99629	0.99818	0.99906
82	0.99859	1	0.99929	0.99886	0.99948
86	0.99859	1	0.99934	0.99929	0.99972
90	0.99859	1	0.99964	0.99957	0.99985
94	0.99859	1	0.99964	0.99974	0.99992
98	0.99859	1	0.99964	0.99985	0.99996

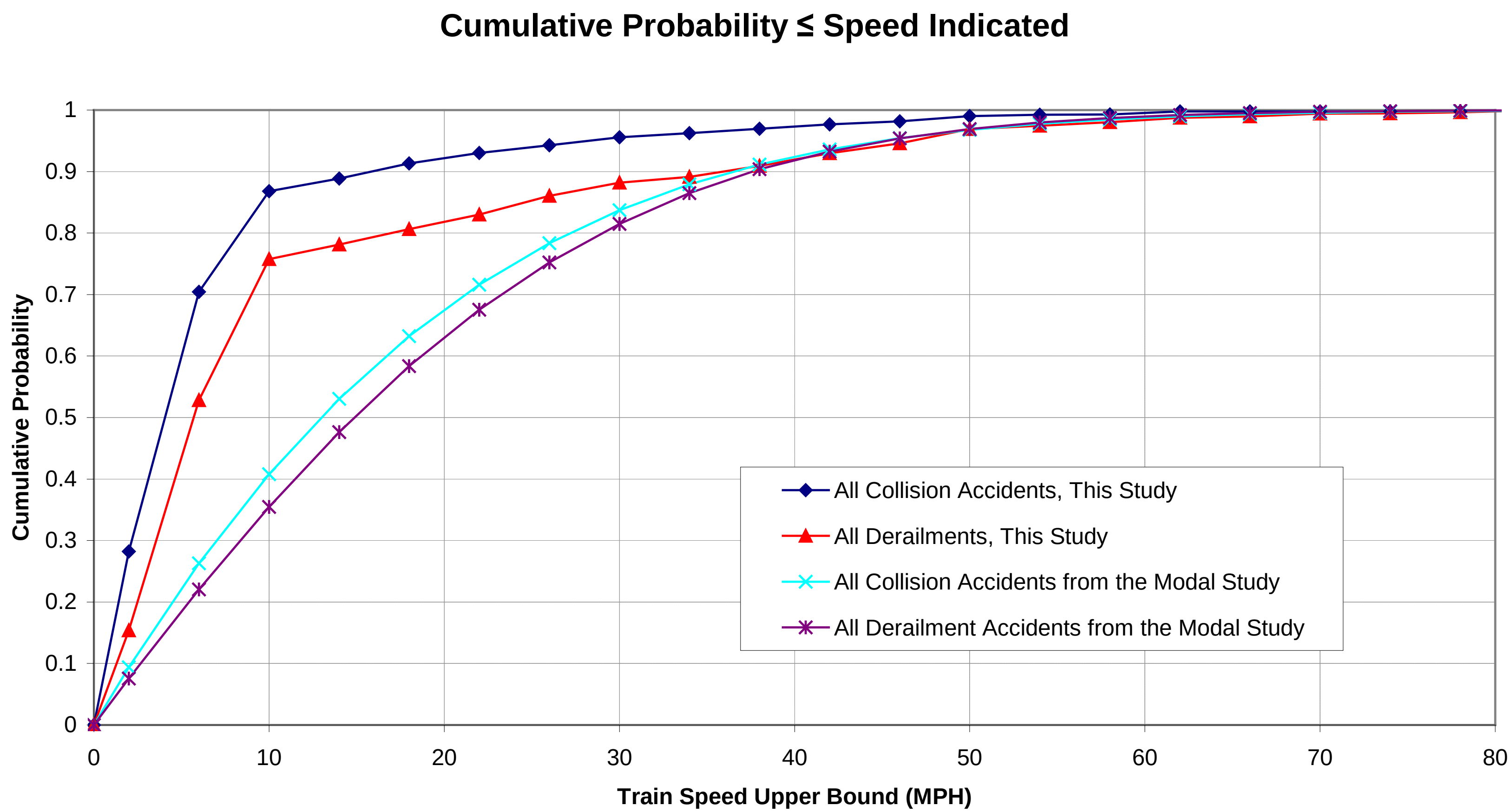

Figure 3-6
Cumulative Probability of Train Accident Speed

3.4.3 Freight Train Accidents Involving HAZMAT

Another key issue that can be analyzed with the FRA database is whether or not a freight train reporting an accident included cars carrying hazardous material (HAZMAT cars) and whether or not HAZMAT cars were damaged as a result of the reported accident. Table 3-8 summarizes freight train accidents with HAZMAT car damage by year for all track classes and speed ranges. The resulting frequency was 3.06E-07 events per freight train-mile.

Table 3-8
Summary of Freight Train Accidents with HAZMAT Car Damage

Year	Number of Accidents	Freight Train-Miles
2000	195	549,382,910
2001	184	537,801,654
2002	184	546,885,449
2003	180	561,202,409
2004	164	583,942,080
2005	140	597,940,859
2006 (through May)	67	258,290,583
TOTAL	1,114	3,635,445,944

Table 3-9 summarizes freight train accidents with HAZMAT car damage and accident speeds of at least 30 MPH. The resulting frequency was 8.45E-08 events per freight train-mile.

Table 3-9
Summary of Freight Train Accidents with HAZMAT Car Damage and Speed ≥ 30 MPH on Track Class 3+

Year	Number of Accidents	Freight Train-Miles
2000	49	493,648,412
2001	51	483,242,066
2002	46	491,404,316
2003	50	504,268,831
2004	36	524,701,579
2005	34	537,280,192
2006 (through May)	10	232,087,191
TOTAL	276	3,266,632,587

Table 3-10 summarizes freight train accident data applied to calculate the frequency of primary and secondary derailment accidents for freight trains consisting of one or more HAZMAT cars and reporting accident speeds of at least 60 MPH. The point estimate for this frequency was 1.05E-08 events per freight train-mile.

Table 3-10
Summary of HAZMAT Freight Train Primary and Secondary Derailment Accidents at Speed ≥ 60 MPH on Track Class 4+

Year	Number of Accidents	Freight Train-Miles
2000	3	358,293,202
2001	6	350,740,209
2002	3	356,664,423
2003	5	366,001,571
2004	3	380,831,791
2005	5	389,961,430
2006 (through May)	0	168,450,380
TOTAL	25	2,370,943,007

Table 3-11 summarizes freight train accident data used to calculate the frequency of accidents in which at least one HAZMAT car experienced damage and reported train speed was at least 60 MPH. The point estimate for this second frequency was calculated to be 8.01E-09 events per freight train-mile. (Trains carrying dangerous or significant quantities of HAZMAT have lower speed limits and additional operational controls placed upon them.)

Table 3-11
Summary of Freight Train Primary and Secondary Derailment Accidents with HAZMAT Car Damage at Speed ≥ 60 MPH on Track Class 4+

Year	Number of Accidents	Freight Train-Miles
2000	2	358,293,202
2001	5	350,740,209
2002	3	356,664,423
2003	5	366,001,571
2004	1	380,831,791
2005	3	389,961,430
2006 (through May)	0	168,450,380
TOTAL	19	2,370,943,007

3.5 Train Accident Risk Assessment

Six bounding and two best-estimate case studies were performed to predict the frequency of SNF cask immersion in water as a result of train accidents. Figures 3-7 through 3-14 show the quantification of each case following the basic structure described in Figure 3-3. The cases progress from the general population of all trains to those judged to be the most representative of the type of train that transport spent nuclear fuel. Table 3-12 presents a summary of the major differentiating characteristics of these eight case studies. Table 3-13 provides a summary of the case study initiating event frequencies derived in Section 3.4.

NUREG/CR-6672 event tree values in Figure 3-2 were applied in all cases for the following parameters:

Both Sequences

- Conditional probability of a derailment, given any accident: 0.8187 (81.87%). Note: this factor is applied only to the accident rates involving all accidents.

Sequence 3

Given a derailment,

- Conditional probability that the accident occurs over a bridge: 0.0097 (0.97%)
- Conditional probability that one or more SNF casks from the accident over the bridge will fall into the water under the bridge: 0.2034 (20.3%)

Sequence 8

Given a derailment,

- Conditional probability that the accident occurs over an embankment: 0.0110 (1.10%)
- Conditional probability that one or more SNF casks from the accident over the embankment will go into a ditch: 0.3812 (38.12%)

The event sequence results are obtained by multiplying the initiating event frequency by the conditional point estimate probability values along each branch of the train accident event tree. The total full immersion frequency for the case (sum of event Sequences 3 and 8 frequency values), and the “percentage of accidents” value are presented for this and each succeeding train accident case analyzed in this project. The “percentage of accidents” value is simply the percentage of the initiating train accident that is estimated to result in SNF cask full immersion.

Table 3-12
Summary of Case Study Differentiating Characteristics

Case Study	All Train Accidents 2000-May 2006	Freight Train Accidents 2000-May 2006	Includes Primary Derail-ments Criterion	Includes Other Derail-ments Criterion	Includes Speed Range ≥ 60 MPH Criterion	Includes Speed Range ≥ 30 MPH Criterion	Track Class Exposure Criterion	Includes HAZMAT Train Criterion	Includes HAZMAT Car Damage Criterion
1	X						All		
2		X					All		
3		X	X	X			All		
4		X				X	3+		
5		X					All		X
6		X				X	3+		X
7		X	X	X	X		4+	X	
8		X	X	X	X		4+		X

Table 3-13
Summary of Initiating Event Frequencies for Risk Analysis Case Studies

Case Study Number	Case Study Initiating Event Description	Point Estimate Frequency (Events/ Train-Mile)
1	All Train Accidents per Train-Mile (All Accidents, All Speeds, All Track Classes), 2000 - May 2006.	4.33E-06
2	Freight Train Accidents per Freight Train-Mile (All Accidents, All Speeds, All Track Classes), 2000 - May 2006.	2.67E-06
3	Freight Train Accidents per Freight Train-Mile (Accidents with Primary or Secondary Derailments, All Speeds, All Track Classes), 2000 - May 2006.	2.25E-06
4	Freight Train Accidents per Track Class 3+ Freight Train-Mile (using Table 2-4 of Ref. 8) with Speed ≥ 30 MPH, 2000 - May 2006.	6.51E-07
5	Freight Train Accidents per Freight Train-Mile (Accidents with HAZMAT Car Damage, All Speeds, All Track Classes), 2000 - May 2006.	3.06E-07
6	Freight Train Accidents per Freight Train-Mile (Accidents with HAZMAT Car Damage, ≥ 30 MPH, Track Class 3+), 2000 - May 2006.	8.45E-08
7	HAZMAT Freight Train Primary and Secondary Derailment Accidents per Track Class 4+ Freight Train-Mile (using Table 2-4 of Ref. 8) with Speed ≥ 60 MPH, 2000 - May 2006.	1.05E-08
8	Freight Train Primary and Secondary Derailment Accidents per Freight Train-Mile (Accidents with HAZMAT Car Damage, ≥ 60 MPH, Track Class 4+), 2000 - May 2006.	8.01E-09

3.5.1 Bounding Assessments

The bounding risk analyses conservatively assume that SNF casks going into a body of water will become totally immersed. This is clearly a bounding assumption, as some streams and rivers are shallow, and many drainage ditches are either too small to allow full immersion, or they contain water only during or after a storm.

Case 1 addresses the accident rates for all trains and all accidents. This duplicates the train accident rate used for both the Modal Study and NUREG/CR-6632.

Frequency of Train Accidents of Interest (Events/Train-Mile)[1]	Cond. Prob. of a Derailment[2]	Cond. Prob. Over Bridge[2]	Cond. Prob. Into Water[2]	Cond. Prob. Of Cask Full Immersion[3]	Frequency Cask Fully Immersed in Water (Events/Train-Mile)
4.33E-06	0.818722	0.0097	0.20339	1	7.0E-09
		Cond. Prob. Over Embankment	**Cond. Prob. Into Drainage Ditch**		
		0.0110	0.3812	1	1.5E-08
				FREQUENCY OF TOTAL FULL IMMERSION	2.2E-08
				PERCENT OF TRAIN ACCIDENTS OF INTEREST	0.5%

[1]All Train Accidents per Train-Mile (All Accidents, All Speeds, All Track Classes), 2000 - May 2006.
[2]From NUREG/CR-6672
[3]Estimated based on engineering judgment

Figure 3-7
Case 1 (All train accidents, all speeds, all track classes, using 2000 - May 2006 data)

It can be seen that the assumption of guaranteed full immersion results in about 1 in 200 train accidents producing full immersion in water, which over estimates the frequency of immersion.

Figure 3-8 shows the results for Case 2, which includes only freight trains.

Frequency of Train Accidents of Interest (Events/Train-Mile)[1]	Cond. Prob. of a Derailment[2]	Cond. Prob. Over Bridge[2]	Cond. Prob. Into Water[2]	Cond. Prob. Of Cask Full Immersion[3]	Frequency Cask Fully Immersed in Water (Events/Train-Mile)
2.67E-06	8.19E-01	0.0097	0.20339	1	4.3E-09
		Cond. Prob. Over Embankment	**Cond. Prob. Into Drainage Ditch**		
		0.011	0.3812	1	9.2E-09
				FREQUENCY OF TOTAL FULL IMMERSION	1.3E-08
				PERCENT OF TRAIN ACCIDENTS OF INTEREST	0.5%

[1]Freight Train Accidents per Freight Train-Mile (All Accidents, All Speeds, All Track Classes), 2000 - May 2006.
[2]From NUREG/CR-6672
[3]Estimated based on engineering judgment

Figure 3-8
Case 2 (All freight train accidents, all speeds, all track classes, using 2000 - May 2006 data)

The lower frequency of freight train accidents is believed to arise from the number of miles that can be accumulated on long haul routes, whereas passenger trains, such as commuter lines, typically have shorter routes in congested regions.

Figure 3-9 shows Case 3, in which the initial accident frequency is restricted to freight train accidents in which at least one car has derailed. Therefore, the conditional probability of a derailment is set to one.

Frequency of Train Accidents of Interest (Events/Train-Mile)[1]	Cond. Prob. of a Derailment	Cond. Prob. Over Bridge[2]	Cond. Prob. Into Water[2]	Cond. Prob. Of Cask Full Immersion[3]	Frequency Cask Fully Immersed in Water (Events/Train-Mile)
2.25E-06	1	0.0097	0.20339	1	4.4E-09
		Cond. Prob. Over Embankment	**Cond. Prob. Into Drainage Ditch**		
		0.011	0.3812	1	9.4E-09
				FREQUENCY OF TOTAL FULL IMMERSION	1.4E-08
				PERCENT OF TRAIN ACCIDENTS OF INTEREST	0.6%

[1]Freight Train Accidents per Freight Train-Mile (Accidents with Primary or Secondary Derailments, All Speeds, All Track Classes), 2000 - May 2006.
[2]From NUREG/CR-6672
[3]Estimated based on engineering judgment

Figure 3-9
Case 3 (Freight train accidents from all primary and secondary derailments, all speeds, all track classes, using 2000-May 2006 data)

The resulting frequency for Case 3 is very close of Case 2. This indicates that the current experience in the FRA database agrees with the conditional probability of derailment cited in the NUREG studies.

Figure 3-10 shows Case 4, which is the first to consider accidents at reasonable transit speeds.

Frequency of Train Accidents of Interest (Events/Train-Mile)[1]	Cond. Prob. of a Derailment[2]	Cond. Prob. Over Bridge[2]	Cond. Prob. Into Water[2]	Cond. Prob. Of Cask Full Immersion[3]	Frequency Cask Fully Immersed in Water (Events/Train-Mile)
6.51E-07	8.19E-01	0.0097	0.20339	1	1.1E-09
		Cond. Prob. Over Embankment	**Cond. Prob. Into Drainage Ditch**		
		0.011	0.3812	1	2.2E-09
				FREQUENCY OF TOTAL FULL IMMERSION	3.3E-09
				PERCENT OF TRAIN ACCIDENTS OF INTEREST	0.5%

[1]Freight Train Accidents per Track Class 3+ Freight Train-Mile (using Table 2-4 of Ref. 8) with Speed = 30 MPH, 2000 - May 2006.
[2]From NUREG/CR-6672
[3]Estimated based on engineering judgment

Figure 3-10
Case 4 [Freight train accidents per Track Class 3+ freight train-mile (using Table 3-4 of Ref. 8) with speed $\geq$ 30 MPH, using 2000 - May 2006 data]

The reduction in the accident frequency by a factor of more than three from Case 2 reflects the fact that most freight train accidents occur at slow speeds, which agrees with the cumulative distributions in Figure 3-6. This is a reasonable expectation, as freight trains undergo a considerable amount of handling in switch yards as the freight cars are being cut from and added to the train’s consist.

Figure 3-11 shows the accident rate of freight trains that included HAZMAT car damage. As experience data regarding the trains that carried HAZMAT without being involved in an accident are not available, overall freight train miles was used to calculate this accident frequency.

Frequency of Train Accidents of Interest (Events/Train-Mile)[1]	Cond. Prob. of a Derailment[2]	Cond. Prob. Over Bridge[2]	Cond. Prob. Into Water[2]	Cond. Prob. Of Cask Full Immersion[3]	Frequency Cask Fully Immersed in Water (Events/Train-Mile)
3.06E-07	8.19E-01	0.0097	0.20339	1	4.9E-10
		Cond. Prob. Over Embankment	**Cond. Prob. Into Drainage Ditch**		
		0.011	0.3812	1	1.1E-09
				FREQUENCY OF TOTAL FULL IMMERSION	1.5E-09
				PERCENT OF TRAIN ACCIDENTS OF INTEREST	0.5%

[1]Freight Train Accidents per Freight Train-Mile (Accidents with HAZMAT Car Damage, All Speeds, All Track Classes), 2000 - May 2006.
[2]From NUREG/CR-6672
[3]Estimated based on engineering judgment

Figure 3-11
Case 5 [Freight train accidents per freight train-mile (Accidents with HAZMAT car damage, all speeds, all track classes), using 2000 - May 2006 data]

Comparison of Case 2 and Case 5 indicates that about 1 in 9 accidents involved some form of HAZMAT car damage.

Frequency of Train Accidents of Interest (Events/Train-Mile)[1]	Cond. Prob. of a Derailment[2]	Cond. Prob. Over Bridge[2]	Cond. Prob. Into Water[2]	Cond. Prob. Of Cask Full Immersion[3]	Frequency Cask Fully Immersed in Water (Events/Train-Mile)
8.45E-08	8.19E-01	0.0097	0.20339	1	1.4E-10
		Cond. Prob. Over Embankment	**Cond. Prob. Into Drainage Ditch**		
		0.011	0.3812	1	2.9E-10
				FREQUENCY OF TOTAL FULL IMMERSION	4.3E-10
				PERCENT OF TRAIN ACCIDENTS OF INTEREST	0.5%

[1]Freight Train Accidents per Freight Train-Mile (Accidents with HAZMAT Car Damage, > 29 MPH, Track Class 3+), 2000 - May 2006.
[2]From NUREG/CR-6672
[3]Estimated based on engineering judgment

Figure 3-12
Case 6 [Freight train accidents per freight train-mile (Accidents with HAZMAT car damage, ≥30 MPH, Track Class 3+), using 2000 - May 2006 data]

Comparison of Case 6 with Case 5 again reflects the fact that most freight train accident occur at slow speeds.

3.5.2 Best Estimate Risk Assessments

The best estimate risk assessments quantify two cases involving HAZMAT accidents at speeds exceeding 60 MPH. These speeds have the kinetic energy to produce > 2% strain damage, considered necessary for ingress of water. Tables 3-9 and 3-10 listed the numbers of accidents involved in Cases 7 and 8, respectively.

For these assessments, the conditional probabilities of full immersion were estimated using engineering judgment.

- The probability of full immersion given the cask falls off a bridge into water is estimated to be 0.8. It is recognized that most large rivers are deep enough in their channels to fully cover a cask, but many smaller rivers and streams can be shallow. The estimate reflects a conservative estimate of the ratio of deep to shallow water in the streams and rivers crossed by railroads.
- The probability of full immersion given the cask lands in a drainage ditch or adjacent body of water is estimated to be 0.05. Not all drainage ditches contain water all the time, and most ditches are too shallow to fully immerse a cask. Adjacent bodies of water could be deep, but generally the water along the shoreline is relatively shallow. These factors combined to justify the assessment.

Case 7 examines all accidents involving HAZMAT trains at speeds exceeding 60 MPH.

Frequency of Train Accidents of Interest (Events/Train-Mile)[1]	**Cond. Prob. of a Derailment**	**Cond. Prob. Over Bridge[2]**	**Cond. Prob. Into Water[2]**	**Cond. Prob. Of Cask Full Immersion[3]**	**Frequency Cask Fully Immersed in Water (Events/Train-Mile)**
1.05E-08	1	0.0097	0.2034	0.80	1.7E-11
		Cond. Prob. Over Embankment	**Cond. Prob. Into Drainage Ditch**		
		0.0110	0.3812	0.05	2.2E-12
				FREQUENCY OF TOTAL FULL IMMERSION	1.9E-11
				PERCENT OF TRAIN ACCIDENTS OF INTEREST	0.2%

[1]HAZMAT Freight Train Primary & Secondary Derailment Accidents per Track Class 4+ Freight Train-Mile, Speed = 60 MPH
[2]From NUREG/CR-6672
[3]Estimated based on engineering judgment

Figure 3-13
Case 7 [HAZMAT freight train primary and secondary derailment accidents per Track Class 4+ freight train-mile (using Table 3-4 of Reference 8) with speed ≥ 60 MPH, all track classes), using 2000 - May 2006 data]

Case 7 illustrates that the accidents involving HAZMAT trains are only a small fraction of all freight train accidents. This is to be expected, as HAZMAT is specifically identified within trains, and controls can be applied if necessary. One such control is to simply lower the allowable speed of a train transporting dangerous HAZMAT.

Case 8 estimates the frequency of accidents with some kind of HAZMAT car damage, both major and minor.

Frequency of Train Accidents of Interest (Events/Train-Mile)[1]	Cond. Prob. of a Derailment	Cond. Prob. Over Bridge[2]	Cond. Prob. Into Water[2]	Cond. Prob. Of Cask Full Immersion[3]	Frequency Cask Fully Immersed in Water (Events/Train-Mile)
8.01E-09	1	0.0097	0.2034	0.80	1.3E-11
		Cond. Prob. Over Embankment	**Cond. Prob. Into Drainage Ditch**		
		0.0110	0.3812	0.05	1.7E-12
				FREQUENCY OF TOTAL FULL IMMERSION	1.4E-11
				PERCENT OF TRAIN ACCIDENTS OF INTEREST	0.2%

[1]Freight Train Accidents per Freight Train-Mile (Primary and Secondary Derailment Accidents with HAZMAT Car Damage, = 60 MPH).
[2]From NUREG/CR-6672
[3]Estimated based on engineering judgment

Figure 3-14
Case 8 [Freight train primary and secondary derailment accidents per freight train-mile (Accidents with HAZMAT car damage, ≥ 60 MPH, Track Class 4+), using 2000 - May 2006 data]

The Case 8 frequency is only 25% lower than Case 7, indicating that for most accidents involving trains at high speed carrying some form of HAZMAT, a large majority of the trains do experience some kind of damage to a HAZMAT car. It is uncertain whether the damage is major or minor, but the high percentage of trains that do experience HAZMAT car damage supports the requirement that trains carrying dangerous or significant amounts of HAZMAT have operating limits imposed upon them.

3.6 Results and Observations

Table 3-14 presents a summary of the point estimate accident of interest (AOI) frequency results for all the case studies analyzed in this project.

Table 3-14
Summary of AOI Frequency Results for All Case Studies

Case Study Number	Case Study Initiating Event Description	Point Estimate AOI Frequency Results (Events/Train-Mile)
1	All Train Accidents per Train-Mile (All Accidents, All Speeds, All Track Classes), 2000 - May 2006.	2.2E-08
2	Freight Train Accidents per Freight Train-Mile (All Accidents, All Speeds, All Track Classes), 2000 - May 2006.	1.3E-08
3	Freight Train Accidents per Freight Train-Mile (Accidents with Primary or Secondary Derailments, All Speeds, All Track Classes), 2000 - May 2006.	1.4E-08
4	Freight Train Accidents per Track Class 3+ Freight Train-Mile (using Table 2-4 of Ref. 8) with Speed ≥ 30 MPH, 2000 - May 2006.	3.3-09
5	Freight Train Accidents per Freight Train-Mile (Accidents with HAZMAT Car Damage, All Speeds, All Track Classes), 2000 - May 2006.	1.5E-09
6	Freight Train Accidents per Freight Train-Mile (Accidents with HAZMAT Car Damage, ≥ 30 MPH, Track Class 3+), 2000 - May 2006.	4.3E-10
7	HAZMAT Freight Train Primary and Secondary Derailment Accidents per Track Class 4+ Freight Train-Mile (using Table 2-4 of Ref. 8) with Speed ≥ 60 MPH, 2000 - May 2006.	1.9E-11
8	Freight Train Primary and Secondary Derailment Accidents per Freight Train-Mile (Accidents with HAZMAT Car Damage, ≥ 60 MPH, Track Class 4+), 2000 - May 2006.	1.4E-11

Based on current FRA train accident and operational data, the results of this study show that the likelihood of SNF cask total immersion in water as a result of SNF cask shipment train accidents is extremely low. The train accident frequency analysis shows that using current FRA data yields lower frequencies for accident initiating events and, thus, for a potential fuel cask criticality accident. Applying the capabilities of the current FRA database structure and relational database techniques, lower accident rates can be predicted for truly relevant train accidents, thereby improving the resolution of the risk assessment as compared to techniques applied in the Modal Study and NUREG/CR-6672.

4
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A
DETAILED HFE REPORTS

HAFMS1, RE Prepare Fuel Movement Sequence Data Sheet

Basic Event Summary

Analyst:	Dykes, AA
Rev. Date:	11/18/08
Reviewer:	Johnson, DH
Cognitive Method:	THERP
Analysis Database:	1850433 EPRI 1016635 Trans Crit HRA.HRA (11/18/08, 516096 Bytes)

Table A-1
HAFMS1 Summary

Analysis Results:	without Recovery	with Recovery
P_{exe}	1.3e-02	1.3e-02
Total HEP		1.3e-02
Error Factor		5

Related Human Interactions

The Fuel Movement Sequence Data Sheet is prepared from the DSC Fuel Loading Pattern Form provided by Fuel Services

Procedures

Performance: (Revision:)

Period of Performance:

Testing: (Revision:)

Period of Testing:

Procedure Notes

EXAMPLE SO23-X-9, Rev 5, Para 3.10

HFE Scenario Description

This action is primarily an administrative function involving identifying the location of F/As with specific S/Ns within the SFP and associating each with the location within the DSC specified by the DSC Fuel Loading Pattern form provided by Nuclear Fuel Services. Locations within the SFP are recorded in plant nuclear fuel inventory software as well as on a tag board.

The refueling engineer generates the transfer sequence using software designed to assist this process and verifies that the resulting Fuel Movement Sequence Data Sheet correctly delineates the transfer sequence.

Performance Shaping Factors

This action is done manually in an office environment.

At the example plant, both the SFP inventory and the layout of the DSC are displayed in graphical format produced by a computer program. The refueling engineer develops the data for the FMSDS by clicking on the storage cell and moving it to the location in the DSC specified by the DSC Fuel Loading Pattern Form.

Although there is no step by step direction for creating the FMSDS, it can be assumed that the refueling engineer is qualified to accomplish the activity without written steps.

Execution Unrecovered

HAFMS1

Table A-2
HAFMS1 Execution Unrecovered

Procedure		Comment				Stress Factor	Over Ride
Step No.	Instruction/Comment	Error Type	THERP		HEP		
			Table	Item			
1	Identify location of F/A S/N on FLP Form for each of 32 F/As	32 chances to make an error = 1.3E-3*32 = 4.2E-02				1	4.2E-02
		EOC	20-10	9	1.3E-3		
	Total Step HEP						4.2e-02
2	Specify the FROM and to locations for the 32 transfers using ShuffleWorks	32 chances to make an error = 1.3E-3*32 = 4.2E-02				1	.042
		EOC	20-10	2	1.3E-3		
	Total Step HEP						4.2e-02
3	Verify that S/N and DSC cell locations match on the printed DSC Fuel Loading Pattern form	Once a DMSD sheet is printed out from the computer, the individual will check it by actively comparing it with the DSC Fuel Loading Pattern form. This is equivalent to a check but is judged to have moderate dependence since one person does both.				1	
		EOC	20-22	4	1.6E-2		
	Total Step HEP						1.6e-02

Execution Recovery

HAFMS1

Table A-3
HAFMS1 Execution Recovery

Critical Step No.	Recovery Step No.	Action	HEP (Crit)	HEP (Rec)	Dep.	Cond. HEP (Rec)	Total for Step
1		**Identify location of F/A S/N on FLP Form for each of 32 F/As**	**4.2e-02**				**6.6e-03**
	3	Verify that S/N and DSC cell locations match on the printed DSC Fuel Loading Pattern form		1.6e-02	MD	1.6e-01	
2		**Specify the FROM and to locations for the 32 transfers using ShuffleWorks**	**4.2e-02**				**6.6e-03**
	3	Verify that S/N and DSC cell locations match on the printed DSC Fuel Loading Pattern form		1.6e-02	MD	1.6e-01	
Total Unrecovered:			**8.4e-02**	**Total Recovered:**			**1.3e-02**

HASEL1, Select F/As Conforming to Certificate of Compliance

Basic Event Summary

Analyst:	Dykes, AA
Rev. Date:	11/18/08
Reviewer:	Johnson, DH
Cognitive Method:	THERP
Analysis Database:	1850433 EPRI 1016635 Trans Crit HRA.HRA (11/18/08, 516096 Bytes)

Table A-4
HASEL1 Summary

Analysis Results:	without Recovery	with Recovery
P_{exe}	2.5e-04	2.5e-04
Total HEP		2.5e-04
Error Factor		10

Related Human Interactions

This action is a Nuclear Fuel Management administrative function to generate the DSC Fuel Loading Pattern Form. The initial enrichment and burnup of every F/A selected for loading must be in conformance with the CoC.

F/A initial enrichment and burnup recorded in plant SNM accountability software. Burnup data can be verified by reference to the plant's core follow and fuel cycle management software.

Procedures

Performance: (Revision :)

Period of Performance:

Testing: (Revision :)

Period of Testing:

Procedure Notes

HFE Scenario Description

This is an administrative action is done in an office environment.

Performance Shaping Factors

This is an administrative action is done in an office environment. The required information can be extracted electronically from the SNM accountability software, but will require manual query and decision activities.

Initial HEP is a nominal human error probability for error of commission when no other human error can be found. The action includes recovery from errors by error of commission during active independent verification that the Fuel Loading Pattern Form is correct.

There is no time limit for accomplishing this action.

Execution Unrecovered

HASEL1

Table A-5
HASEL1 Execution Unrecovered

Procedure		Comment				Stress Factor	Over Ride
Step No.	**Instruction/Comment**	**Error Type**	**THERP**		**HEP**		
			Table	**Item**			
1	Error of Commission during selection of F/As for DSC load	Based on CAL-WHS-MD-000003 REV OOA, Table 2, Item 1, Nominal human error probability for error of commission when no other human error can be found in the Tables. The median value is 3E-03 and the range factor of uncertainty is 5. Source: NUREG 1278, HEP is found on page G-. Uncertainty factor Table 20-20, Item 7.				1	3.9E-03
	Total Step HEP						3.9e-03
2	Independently verify selection					1	
		EOC	20-22	4	1.6E-2		
	Total Step HEP						1.6e-02

Execution Recovery

HASEL1

Table A-6
HASEL1 Execution Recovery

Critical Step No.	Recovery Step No.	Action	HEP (Crit)	HEP (Rec)	Dep.	Cond. HEP (Rec)	Total for Step
1		**Error of Commission during selection of F/As for DSC load**	**3.9e-03**				**2.5e-04**
	2	Independently verify selection		1.6e-02	LD	6.5e-02	
Total Unrecovered:			**3.9e-03**	**Total Recovered:**			**2.5e-04**

HRDSC1, Visually Verify all F/A S/N against DSC Fuel Loading Pattern

Basic Event Summary

Analyst:	Dykes, AA
Rev. Date:	11/18/08
Reviewer:	Johnson, DH
Cognitive Method:	THERP
Analysis Database:	1850433 EPRI 1016635 Trans Crit HRA.HRA (11/18/08, 516096 Bytes)

Table A-7
HRDSC1 Summary

Analysis Results:	without Recovery	with Recovery
P_{exe}	2.8e-03	2.8e-03
Total HEP		2.8e-03
Error Factor		5

Related Human Interactions

Procedures

Performance: (Revision:)

Period of Performance:

Testing: (Revision:)

Period of Testing:

Procedure Notes

SO23-X-7.2, para. 2.2.3

HFE Scenario Description

Representative from Nuclear Oversight verifies with the refueling engineer via three-way communication the location and serial number of fuel assemblies in the DSC while videotaping DSC contents

Performance Shaping Factors

Operations are performed in the SFP with no time limits or distractions. Work can be done within one shift, and the qualifications of all operators are verified.

Execution Unrecovered

HRDSC1

Table A-8
HRDSC1 Execution Unrecovered

Procedure		Comment				Stress Factor	Over Ride
Step No.	**Instruction/Comment**	**Error Type**	**THERP**		**HEP**		
			Table	**Item**			
1	RE states DSC Location and Serial Number to Nuc Oversight Rep.	Error rate = 1.3E-03*32 = 8.4E-02 based on 32 F/As in the DSC				1	8.4E-02
		EOC	20-10	9	1.3E-3		
	Total Step HEP						8.4e-02
2	Nuc Oversight Rep views DSC location and F/A SN and states what he sees					1	
		EOC	20-22	4	1.6E-2		
	Total Step HEP						1.6e-02
3	RE verifies both stated items with "That is correct"	Two separate items = 2*1.6E-02 = 3.2E-02				1	3.2E-02
		EOC	20-22	4	1.6E-2		
	Total Step HEP						3.2e-02

Execution Recovery

HRDSC1

Table A-9
HRDSC1 Execution Recovery

Critical Step No.	Recovery Step No.	Action	HEP (Crit)	HEP (Rec)	Dep.	Cond. HEP (Rec)	Total for Step
1		**RE states DSC Location and Serial Number to Nuc Oversight Rep.**	**8.4e-02**				**2.8e-03**
	2	Nuc Oversight Rep views DSC location and F/A SN and states what he sees		1.6e-02	LD	6.5e-02	
	3	RE verifies both stated items with "That is correct"		3.2e-02	HD	5.2e-01	
Total Unrecovered:			**8.4e-02**	**Total Recovered:**			**2.8e-03**

HRDSC2, Independent Verification of Spent Fuel S/Ns in the DSC

Basic Event Summary

Analyst:	Dykes, AA
Rev. Date:	11/18/08
Reviewer:	Johnson, DH
Cognitive Method:	THERP
Analysis Database:	1850433 EPRI 1016635 Trans Crit HRA.HRA (11/18/08, 516096 Bytes)

Table A-10
HRDSC2 Summary

Analysis Results:	without Recovery	with Recovery
P_{exe}	7.4e-02	7.4e-02
Total HEP		7.4e-02
Error Factor		5

Related Human Interactions

Done after DSC is fully loaded and visual verification of F/A S/N has been completed. No further activities associated with the DSC may be accomplished until the independent verification is completed.

Procedures

Performance: (Revision:)

Period of Performance:

Testing: (Revision:)

Period of Testing:

Procedure Notes

EXAMPLE SO23-X-9, Rev 5, Para 3.10

HFE Scenario Description

A qualified independent person reviews the video recording and verifies the S/N and location of F/As within the DSC against the DSC Fuel Loading Pattern form provided by Nuclear Fuel Management.

Performance Shaping Factors

The review of the video tape can be done in an office environment. The reviewer observes the S/N and location of each of 32 F/As on the video and compares it to the DSC Fuel Loading Pattern form provided by Nuclear Fuel Services.

Although there is no step by step direction for accomplishing the transfer, it can be assumed that the reviewer is qualified to accomplish the activity without written steps. As this action is done by one person, the only recovery from an error is self checking.

Execution Unrecovered

HRDSC2

Table A-11
HRDSC2 Execution Unrecovered

Procedure		Comment				Stress Factor	Over Ride
Step No.	**Instruction/Comment**	**Error Type**	**THERP**		**HEP**		
			Table	**Item**			
1	Review location of F/A S/N on FLP Form for each of 32 F/As	32 chances to make an error = 5.1E-3*0.5 *32= 4.2E-02. The factor of 0.5 takes credit for fuel management checkin maintenance (Table 20-22, Item 10).				1	8.2E-02
	--	EOM	20-7	2	3.8E-3		
		EOC	20-10	9	1.3E-3		
	Total Step HEP						8.2e-02
2	Self check the verification process prior to signing form	Once the transfer form is printed out from the computer, an individual will check it by actively comparing it with the DSC Fuel Loading Pattern form. This constitutes an active check using a new document requiring a verification signature.				1	
		EOC	20-22	8	8.1E-1		
	Total Step HEP						8.1e-01

Execution Recovery

HRDSC2

Table A-12
HRDSC2 Execution Recovery

Critical Step No.	Recovery Step No.	Action	HEP (Crit)	HEP (Rec)	Dep.	Cond. HEP (Rec)	Total for Step
1		**Review location of F/A S/N on FLP Form for each of 32 F/As**	**8.2e-02**				**7.4e-02**
	2	Self check the verification process prior to signing form		8.1e-01	HD	9.1e-01	
Total Unrecovered:			**8.2e-02**	**Total Recovered:**			**7.4e-02**

HRDSC3, Independent Three-way Comm. Verification of DSC Video against DSC Fuel Loading Pattern Form

Basic Event Summary

Analyst:	Dykes, AA
Rev. Date:	11/18/08
Reviewer:	Johnson, DH
Cognitive Method:	THERP
Analysis Database:	1850433 EPRI 1016635 Trans Crit HRA.HRA (11/18/08, 516096 Bytes)

Table A-13
HRDSC3 Summary

Analysis Results:	without Recovery	with Recovery
P_{exe}	6.8e-04	6.8e-04
Total HEP		6.8e-04
Error Factor		10

Related Human Interactions

Done after DSC is fully loaded and visual verification of F/A S/N has been completed. No further activities associated with the DSC may be accomplished until the independent verification is completed.

Procedures

Performance: (Revision:)

Period of Performance:

Testing: (Revision:)

Period of Testing:

Procedure Notes

SO23-X-9, para. 6.4.5

HFE Scenario Description

A qualified independent person views the video and records the S/N and location of each F/A in the DSC on a blank DSC Load Pattern Form. He then verifies the completed form against the original with a second person having the original DSC Load Pattern Form via three-way communication.

Performance Shaping Factors

The review of the video tape can be done in an office environment.

Active recording of the S/Ns and three-way communication provides a mechanism for verifying the S/Ns and locations by active participation of two people.

Execution Unrecovered

HRDSC3

Table A-14
HRDSC3 Execution Unrecovered

Procedure		Comment				Stress Factor	Over Ride
Step No.	**Instruction/Comment**	**Error Type**	**THERP**		**HEP**		
			Table	**Item**			
1	Observe F/A S/N on video and record on blank DSC FLP form	Error rate = 1.3E-03*32 = 4.2E-02 based on 32 F/As in the DSC				1	4.2E-02
		EOC	20-10	9	1.3E-3		
	Total Step HEP						4.2e-02
2	State filled out DSC Pattern S/N and location with individual having original DSC pattern form	Error rate = 1.3E-03*32 = 4.2E-02 based on 32 F/As in the DSC				1	4.2e-02
		EOC	20-10	9	1.3E-3		
	Total Step HEP						4.2e-02
3	Second ind verify S/N by stating S/N and Location on NFS DSC FLP form	Error rate = 1.3E-03*32 = 4.2E-02 based on 32 F/As in the DSC				1	4.2E-02
		EOC	20-10	9	1.3E-3		
	Total Step HEP						4.2e-02

Execution Recovery

HRDSC3

Table A-15
HRDSC3 Execution Recovery

Critical Step No.	Recovery Step No.	Action	HEP (Crit)	HEP (Rec)	Dep.	Cond. HEP (Rec)	Total for Step
1		**Observe F/A S/N on video and record on blank DSC FLP form**	**4.2e-02**				**6.8e-04**
	2	State filled out DSC Pattern S/N and location with individual having original DSC pattern form		4.2e-02	LD	9.0e-02	
	3	Second ind verify S/N by stating S/N and Location on NFS DSC FLP form		4.2e-02	MD	1.8e-01	
Total Unrecovered:			**4.2e-02**	**Total Recovered:**			**6.8e-04**

HRFMS1, Nuclear Fuel Management Verifies S/N and "TO" locations of F/As on FMSDS

Basic Event Summary

Analyst:	Dykes, AA
Rev. Date:	11/18/08
Reviewer:	Johnson, DH
Cognitive Method:	THERP
Analysis Database:	1850433 EPRI 1016635 Trans Crit HRA.HRA (11/18/08, 516096 Bytes)

Table A-16
HRFMS1 Summary

Analysis Results:	without Recovery	with Recovery
P_{exe}	6.6e-02	6.6e-02
Total HEP		6.6e-02
Error Factor		5

Related Human Interactions

Independent verification of correct preparation of the Fuel Movement Sequence Data Sheet from the DSC Fuel Loading Pattern form

Procedures

Performance: (Revision:)

Period of Performance:

Testing: (Revision:)

Period of Testing:

Procedure Notes

EXAMPLE SO23-X-9, Rev 5, Para 3.10

HFE Scenario Description

This action is primarily an administrative function involving verifying the S/N and location of F/As listed on the FMSDS against those specified by the DSC Fuel Loading Pattern Form provided by Nuclear Fuel Management.

The Supervisor, Nuclear Fuel Management, or a designee, compares the FMSDS with the DSC Fuel Loading Pattern Form.

Performance Shaping Factors

This action is done in an office environment comparing two documents side by side.

This is a manual activity requiring careful comparison of 32 individual serial number and locations. However, all of them have a distinct physical meaning to the reviewer, and they have been subject to calculations to produce a Certificate of Compliance for the NRC.

Execution Unrecovered

HRFMS1

Table A-17
HRFMS1 Execution Unrecovered

Procedure		Comment				Stress Factor	Over Ride
Step No.	Instruction/Comment	Error Type	THERP		HEP		
			Table	Item			
1	Review location of F/A S/N on FLP Form for each of 32 F/As	32 chances to make an error = 5.1E-3*0.5 *32= 4.2E-02. The factor of 0.5 takes credit for fuel management checkin maintenance (Table 20-22, Item 10).				1	8.2E-02
	--	EOM	20-7	2	3.8E-3		
		EOC	20-10	9	1.3E-3		
	Total Step HEP						8.2e-02
2	Check the verification process prior to signing form	Once the transfer form is printed out from the computer, an individual will check it by actively comparing it with the DSC Fuel Loading Pattern form. This constitutes an active check using a new document requiring a verification signature.				1	
		EOC	20-22	8	8.1E-1		
	Total Step HEP						8.1e-01

Execution Recovery

HRFMS1

Table A-18
HRFMS1 Execution Recovery

Critical Step No.	Recovery Step No.	Action	HEP (Crit)	HEP (Rec)	Dep.	Cond. HEP (Rec)	Total for Step
1		**Review location of F/A S/N on FLP Form for each of 32 F/As**	**8.2e-02**				**6.6e-02**
	2	Check the verification process prior to signing form		8.1e-01	ZD	8.1e-01	
Total Unrecovered:			**8.2e-02**	**Total Recovered:**			**6.6e-02**

HRSEL1, Perform Audit of F/A Enrichment and Burnup Prior to Shipment

Basic Event Summary

Analyst:	Dykes, AA
Rev. Date:	11/18/08
Reviewer:	Johnson, DH
Cognitive Method:	THERP
Analysis Database:	1850433 EPRI 1016635 Trans Crit HRA.HRA (11/18/08, 516096 Bytes)

Table A-19
HRSEL1 Summary

Analysis Results:	without Recovery	with Recovery
P_{exe}	1.0e-02	1.0e-02
Total HEP		1.0e-02
Error Factor		5

Related Human Interactions

Procedures

Performance: (Revision:)

Period of Performance:

Testing: (Revision:)

Period of Testing:

Procedure Notes

HFE Scenario Description

The action involves active independent verification of both the serial number and initial enrichment and burnup of all the fuel assemblies in a DSC prior to shipping.

Performance Shaping Factors

This is an administrative action is done in an office environment. The required information is extracted from the SNM accountability software, but will require manual query and comparison activities.

Execution Unrecovered

HRSEL1

Table A-20
HRSEL1 Execution Unrecovered

Procedure		**Comment**				**Stress Factor**	**Over Ride**
Step No.	**Instruction/Comment**	**Error Type**	**THERP**		**HEP**		
			Table	**Item**			
1	Failure to Initiate Audit Initial Enrichment and burnup of F/As prior to shipping SFC					1	
	--	EOM	20-7b	1	4.3E-4		
	Total Step HEP						4.3e-04
2	Failure to detect error in selection during check against SNM database					1	
		EOC	99	1	1.0E-2		
	Total Step HEP						1.0e-02

Execution Recovery

HRSEL1

Table A-21
HRSEL1 Execution Recovery

Critical Step No.	Recovery Step No.	Action	HEP (Crit)	HEP (Rec)	Dep.	Cond. HEP (Rec)	Total for Step
1		Failure to Initiate Audit Initial Enrichment and burnup of F/As prior to shipping SFC	4.3e-04				
2		Failure to detect error in selection during check against SNM database	1.0e-02				
Total Unrecovered:			1.0e-02	Total Recovered:			1.0e-02

B
RAIL EQUIPMENT ACCIDENT/INCIDENT

Rail Equipment Accident/Incident Form F 6180.54 Accident Downloads on Demand Data File Structure and Field Input Specifications

RAIL EQUIPMENT

ACCIDENT/INCIDENT

FORM F 6180.54

ACCIDENT DOWNLOADS ON DEMAND

DATA FILE STRUCTURE
AND
FIELD INPUT SPECIFICATIONS

Accident Data On Demand

Rail Equipment Accident/Incident
Accident Downloads on Demand
Data File Structure and Field Input Specifications

FIELD NAME	FILE POSITION	FIELD TYPE	FIELD LENGTH	DEFINITION	BLOCK # ON FORM 6180.54	CONVERSION
amtrak	1	A/N	1	amtrak involvement		
iyr	2 - 3	A/N	2	year of incident	5	
imo	4 - 5	A/N	2	month of incident	5	
railroad	6 - 9	A	4	railroad code (Reporting RR)	1a	
incdtno	10 - 19	A/N	10	railroad assigned number	1b	
iyr2	20 - 21	A/N	2	year of incident	5	
imo2	22 - 23	A/N	2	month of incident	5	
rr2	24 - 27	A	4	railroad code (Other RR involved)	2a	
incdtno2	28 - 37	A/N	10	other railroad assigned number	2b	
iyr3	38 - 39	A/N	2	year of incident	5	
imo3	40 - 41	A/N	2	month of incident	5	
rr3	42 - 45	A	4	railroad code (RR responsible for track maintenance)	3a	
incdtno3	46 - 55	A/N	10	RR assigned number	3b	
dummy1	56 - 59	A/N	4	blank data expansion field		
gxid	60 - 66	A/N	7	grade crossing id number	4	
year	67 - 68	A/N	2	year of accident / incident	5	
month	69 - 70	A/N	2	month of incident	5	
day	71 - 72	A/N	2	day of incident	5	
timehr	73 - 74	N	2	hour of incident	6	
timemin	75 - 76	N	2	minute of incident	6	
ampm	77 - 78	A/N	2	am or pm	6	

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Accident Data On Demand

Rail Equipment Accident/Incident
Accident Downloads on Demand
Data File Structure and Field Input Specifications

FIELD NAME	FILE POSITION	FIELD TYPE	FIELD LENGTH	DEFINITION	BLOCK # ON FORM 6180.54	CONVERSION
type	79 - 80	A/N	2	type of accident: 01= derailment 02= head on collision 03= rearend collision 04= side collision 05= raking collision 06= broken train collision 07= hwy-rail crossing 08= RR Grade Crossing 09= obstruction 10= explosive – detonation 11= fire / violent rupture 12= other impacts 13= other (described in narrative)	7	
cars	81 - 83	N	3	# of cars carrying hazmat	8	
carsdmg	84 - 86	N	3	# of hazmat cars damaged or derailed	9	
carshzd	87 - 89	N	3	# of cars that released hazmat	10	
evacuate	90 - 95	N	6	# of persons evacuated	11	
division	96 - 115	A/N	20	railroad division	12	
station	116 - 135	A/N	20	nearest city and town	13	
milepost	136 - 141	A/N	6	milepost #	14	
state	142 - 143	A/N	2	FIPS State code	15	
temp	144 - 146	N	3	temperature in degrees fahrenheit	17	
visiblty	147	A/N	1	daylight period: 1=dawn 2=day 3=dusk 4=dark	18	
weather	148	A/N	1	weather conditions: 1=clear 2=cloudy 3=rain 4=fog 5=sleet 6=snow	19	
trnspd	149 - 151	A/N	3	speed of train in miles per hour: blank=unknown	28	
typspd	152	A/N	1	train speed type: E=estimated R=recorded blank=unknown	28	
trnnbr	153 - 156	A/N	4	train id number	27	
trndir	157	A/N	1	train direction: 1=north 2=south 3=east 4=west	24	
tons	158 - 162	N	5	gross tonnage, excluding power units	29	

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Accident Data On Demand

Rail Equipment Accident/Incident
Accident Downloads on Demand
Data File Structure and Field Input Specifications

FIELD NAME	FILE POSITION	FIELD TYPE	FIELD LENGTH	DEFINITION	BLOCK # ON FORM 6180.54	CONVERSION
typeq	163	A/N	1	type of consist: 1=freight train 2=passenger train 3=commuter train 4=work train 5=single car 6= cut of cars 7= yard / switching 8= light loco(s) 9= maint / inspect car A= spec. MoW eq.	25	Added 'A' to selection options
eqatt	164	A/N	1	equipment attended: 1=yes 2=no	26	
trkname	165 - 184	A/N	20	track identification	21	
trkclas	185	A/N	1	FRA track class: 1-9,X	22	Extended Track class to 9
trkdnsty	186 - 191	A/N	6	annual track density - gross tonnage in millions	23	
typtrk	192	A/N	1	type of track: 1=main 2=yard 3=siding 4=industry	20	
rrcar1	193 - 196	A/N	4	car initials (first involved)	31a(1)	
carnbr1	197 - 202	A/N	6	car number (first involved)	31a(1)	
positon1	203 - 205	A/N	3	car position in train (first involved)	31b(1)	
loaded1	206	A/N	1	car loaded or not (first involved): Y=yes N=no blank=not applicable	31c(1)	
rrcar2	207 - 210	A/N	4	car initials (causing)	31a(2)	
carnbr2	211 - 216	A/N	6	car number (causing)	31a(2)	
positon2	217 - 219	A/N	3	car position in train (causing)	31b(2)	
loaded2	220	A/N	1	car loaded or not (causing): Y=yes N=no blank=not applicable	31c(2)	
headend1	221	N	1	# of head end locomotives	34a(1)	
midman1	222	N	1	# of mid train locomotives, manual	34b(1)	
midrem1	223	N	1	# of mid train locomotives, remote	34c(1)	
rman1	224	N	1	# of rear end locomotives, manual	34d(1)	
rrem1	225	N	1	# of rear end locomotives, remote	34e(1)	
headend2	226	N	1	# of head end locomotives, derailed	34a(2)	
midman2	227	N	1	# of mid train locomotives, manual, derailed	34b(2)	

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Indicates links, changes, or new data

Accident Data On Demand

Rail Equipment Accident/Incident
Accident Downloads on Demand
Data File Structure and Field Input Specifications

FIELD NAME	FILE POSITION	FIELD TYPE	FIELD LENGTH	DEFINITION	BLOCK # ON FORM 6180.54	CONVERSION
midrem2	228	N	1	# of mid train locomotives, remote, derailed	34c(2)	
rman2	229	N	1	# of rear end locomotives, manual, derailed	34d(2)	
rrem2	230	N	1	# of rear end locomotives, remote, derailed	34e(2)	
loadf1	231 - 233	N	3	# of loaded freight cars	35a(1)	
loadp1	234 - 236	N	3	# of loaded passenger cars	35b(1)	
emptyf1	237 - 239	N	3	# of empty freight cars	35c(1)	
emptyp1	240 - 242	N	3	# of empty passenger cars	35d(1)	
caboose1	243 - 245	N	3	# of cabooses	35e(1)	
loadf2	246 - 248	N	3	# of derailed loaded freight cars	35a(2)	
loadp2	249 - 251	N	3	# of derailed loaded passenger cars	35b(2)	
emptyf2	252 - 254	N	3	# of derailed empty freight cars	35c(2)	
emptyp2	255 - 257	N	3	# of derailed empty passenger cars	35d(2)	
caboose2	258 - 260	N	3	# of derailed cabooses	35e(2)	
eqpdmg	261 - 267	N	7	reportable equipment damage in $	36	
trkdmg	268 - 274	N	7	track, signal, way & structure damage in $	37	
cause	275 - 278	A/N	4	primary cause of incident (refer to Appendix C)	38	
cause2	279 - 282	A/N	4	contributing cause of incident (refer to Appendix C)	39	
caskldrr	283 - 285	N	3	# killed for reporting RR - calculated from Form F6180.55a's submitted		
casinjrr	286 - 289	N	4	# injured for reporting RR - calculated from Form F6180.55a's submitted		
caskld	290 - 292	N	3	total killed for all RRs involved - calculated from Form F6180.55a's submitted		
casinj	293 - 296	N	4	total injured for all RRs involved - calculated from Form F6180.55a's submitted		
accause	297 - 300	A/N	4	accident cause code from jointcd 1 record for this incident (refer to Appendix C)		
acctrk	301	A/N	1	type track code from jointcd 1 record for this incident		

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Indicates links, changes, or new data

Accident Data On Demand

Rail Equipment Accident/Incident
Accident Downloads on Demand
Data File Structure and Field Input Specifications

FIELD NAME	FILE POSITION	FIELD TYPE	FIELD LENGTH	DEFINITION	BLOCK # ON FORM 6180.54	CONVERSION
acctrkcl	302	A/N	1	FRA track class from jointcd 1 record for this incident (FRA track class: 1-9,X)		
highspd	303 - 305	A/N	3	maximum speed reported for equipment involved: blank=unknown		
accdmg	306 - 313	N	8	total reportable damage on all reports in $		
dummy2	314 - 316	A/N	3	blank data expansion field		
stcnty	317 - 322	A/N	6	FIPS State & County code		
totinj	323 - 326	N	4	total injured for railroad as reported on Form F6180.54		
dummy3	327 - 332	A/N	6	blank data expansion field		
totkld	333 - 336	N	4	total killed for railroad as reported on Form F6180.54		
engrs	337	A/N	1	# of engineers on duty: blank=not applicable	40	
firemen	338	A/N	1	# of firemen on duty: blank=not applicable	41	
conductr	339	A/N	1	# of conductors on duty: blank=not applicable	42	
brakemen	340	A/N	1	# of brakemen on duty: blank=not applicable	43	
enghr	341 - 342	A/N	2	# of hours engineers on duty: blank=not applicable	44	
engmin	343 - 344	A/N	2	# of minutes engineers on duty (+enghr): blank=not applicable	44	
cdtrhr	345 - 346	A/N	2	# of hours conductors on duty: blank=not applicable	45	
cdtrmin	347 - 348	A/N	2	# of minutes conductors on duty (+cdtrhr): blank=not applicable	45	
jointcd	349	A/N	1	indicates railroad reporting		
region	350	A/N	1	FRA designated region		
dummy4	351	A/N	1	blank data expansion field		
typrr	352 - 353	A/N	2	type railroad - ICC categories; 1st position indicates class 1,2,or 3 RR		

Indicates links, changes, or new data

Accident Data On Demand

Rail Equipment Accident/Incident
Accident Downloads on Demand
Data File Structure and Field Input Specifications

FIELD NAME	FILE POSITION	FIELD TYPE	FIELD LENGTH	DEFINITION	BLOCK # ON FORM 6180.54	CONVERSION
dummy5	354 - 356	A/N	3	blank data expansion field		
rrdiv	357 - 362	A/N	6	RR division code		
method	363 - 382	A/N	20	Method of operation (Series of 1 position codes): A= ATCS B=auto train control C= auto train stop D= cab signals E= traffic control F= interlocking G= automatic block rules H= current of traffic I= time table / train orders J= track warrant control K= direct traffic Control L= yard limits M= special instructions N= other than main track O= other (specify in a narrative) P= positive train control	30	Added 'P' to selection options
narrlen	383 - 386	N	4	length of narrative		
dummy6	387 - 390	A/N	4	blank data expansion field		
year4	391 - 394	A/N	4	Four character year identification		
rrempkld	395 - 397	N	3	# of RR employees killed as reported on Form F6180.54	46	
rrempinj	398 - 400	N	3	# of RR employees injured as reported on Form F6180.54	46	
passkld	401 - 403	N	3	# of passengers killed as reported on Form F6180.54	47	
passinj	404 - 406	N	3	# of passengers injured as reported on Form F6180.54	47	
otherkld	407 - 409	N	3	# of others killed as reported on Form F6180.54	48	
otherinj	410 - 412	N	3	# of others injured as reported on Form F6180.54	48	
county	413 - 432	A/N	20	County Name (See FIPS Codes for associated codes)	16	
cntycd	433 - 435	A/N	3	FIPS County Code		
alcohol	436 - 437	A/N	2	# of positive alcohol tests	32	
drug	438 - 439	A/N	2	# of positive drug tests	32	
dummy7	440 - 451	A/N	12	blank data expansion field		
passtrn	452	A/N	1	were there passengers being transported: Y=yes N=no blank=not applicable	33	
ssb1	453 - 472	A/N	20	special study block 1	49	

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Indicates links, changes, or new data

Accident Data On Demand

Rail Equipment Accident/Incident
Accident Downloads on Demand
Data File Structure and Field Input Specifications

FIELD NAME	FILE POSITION	FIELD TYPE	FIELD LENGTH	DEFINITION	BLOCK # ON FORM 6180.54	CONVERSION
ssb2	473 - 492	A/N	20	special study block 2	49	
narr1	493 - 592	A/N	100	narrative	52	
narr2	593 - 692	A/N	100	narrative	52	
narr3	693 - 792	A/N	100	narrative	52	
narr4	793 - 892	A/N	100	narrative	52	
narr5	893 - 992	A/N	100	narrative	52	
narr6	993 - 1092	A/N	100	narrative	52	
narr7	1093 - 1192	A/N	100	narrative	52	
narr8	1193 - 1292	A/N	100	narrative	52	
narr9	1293 - 1392	A/N	100	narrative	52	
narr10	1393 - 1492	A/N	100	narrative	52	
narr11	1493 - 1592	A/N	100	narrative	52	
narr12	1593 - 1692	A/N	100	narrative	52	
narr13	1693 - 1792	A/N	100	narrative	52	
narr14	1793 - 1892	A/N	100	narrative	52	
narr15	1893 - 1992	A/N	100	narrative	52	
rcl	1993 - 1993	A/N	1	Remote control locomotive = 0,1,2, or 3 0= not a remotely controlled operation 1= remote control portable transmitter 2= remote control tower operation 3= remote control portable transmitter (more than one remote control)	30a	new data
latitude	1994 - 2003	N	10	Latitude in decimal degrees, explicit decimal, explicit +/- (WGS84)	50	new data
longitud	2004 - 2014	N	11	Longitude in decimal degrees, explicit decimal, explicit +/- (WGS84)	51	new data

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Indicates links, changes, or new data

C
RAILROAD INJURY AND ILLNESS SUMMARY

Railroad
Injury and Illness
Summary

FORM F 6180.55

DATA FILE STRUCTURE
AND
FIELD INPUT SPECIFICATIONS

Download File(s)

Railroad Injury and Illness Summary Data File Structure and Field Input Specifications

	FIELD NAME	FILE POSITION	FIELD TYPE	FIELD LENGTH	DEFINITION	BLOCK # ON FORM 6180.55	CONVERSION
√	railroad	1 - 4	A	4	Railroad Code	2	
√	iyr	5 - 6	A/N	2	year of report	3	
√	imo	7 - 8	A/N	2	month of report	3	
√	state	9 - 10	A/N	2	GSA State code (where report was notarized)	4	
√	county	11 - 30	A/N	20	GSA County name (where report was notarized)	5	
	locomi	31 - 40	N	10	locomotive train miles for month		old = data new = zeros
	mtmi	41 - 50	N	10	motor train miles for month		old = data new = zeros
	ysmi	51 - 60	N	10	yard switching miles for month	13	
√	totmi	61 - 70	N	10	total miles as reported on F6180.55		
√	emphrs	71 - 80	N	10	employee man-hours for month	15	
	passmi	81 - 90	N	10	passenger miles for month	16	
	revpass	91 - 100	N	10	# of passengers transported during month	17	
	typrr	101 - 102	A/N	2	type railroad - ICC categories; 1st position indicates class 1,2,or 3 RR		
	region	103	A/N	1	FRA designated region		
	dummy	104	A/N	1	blank data expansion field		
	year4	105 - 108	A/N	4	4 digit year of report	3	
	frtrnmi	109 - 118	N	10	# of train miles during month in freight service	11	old = zeros new = data
	pastrnmi	119 - 128	N	10	# of train miles during month in passenger service	12	old = zeros new = data
	othermi	129 - 138	N	10	any other train miles not included in freight, passenger or yard	14	old = zeros new = data
	cntycd	139 - 141	A/N	3	GSA County code (where report was notarized)		
	stcnty	142 - 147	A/N	6	GSA State and County code (where report was notarized)		
	narr1	148 - 247	A/N	100	narrative	20	old = blank new = data
	narr2	248 - 347	A/N	100	narrative	20	old = blank new = data
	narr3	348 - 397	A/N	50	narrative	20	old = blank new = data

Download File(s)

Railroad Injury and Illness Summary Data File Structure and Field Input Specifications

FIELD NAME	FILE POSITION	FIELD TYPE	FIELD LENGTH	DEFINITION	BLOCK # ON FORM 6180.55	CONVERSION
narrlen	398 - 401	N	4	length of narrative		old = zeros new = data
dummy	402 - 407	A/N	6	blank data expansion field		

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