UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

August 13, 2002

ORGANIZATION: NUCLEAR ENERGY INSTITUTE (NEI)

SUBJECT: SUMMARY OF MEETING WITH THE NEI ON THE INTERNATIONAL NUCLEAR EVENT SCALE (INES) RATINGS

On July 26, 2002, the U. S. Nuclear Regulatory Commission (NRC) Staff met with the NEI in Rockville, Maryland, to discuss the NRC's increased participation in the INES program. This meeting was requested by NEI following issuance of NRC Regulatory Issue Summary (RIS) 2002-01, "Changes to NRC Participation in the International Nuclear Event Scale" dated January 14, 2002. This RIS notified industry of the NRC's increased participation in the INES program.

Background

Participation in the INES program promotes consistent communication regarding the safety significance of reported events at nuclear installations by defining a framework and common terminology for describing events to the nuclear community, the media, and the public. The INES was designed and developed by an international group of experts convened jointly by the International Atomic Energy Agency (IAEA) and the Nuclear Energy Agency (NEA) of the Organization for Economic Cooperation and Development (OECD). The group was guided in its work by the findings of a series of international meetings held to discuss general principles underlying such a scale. The INES also reflects the experience gained from the use of similar scales in France and Japan, as well as consideration of possible scales in several other countries.

The INES classifies events at several levels. Events of greater safety significance (levels 4-7) are termed "accidents," while events of lesser safety significance (levels 1-3) are termed "incidents," and those of no safety significance (level 0 or below) are termed "deviations." A description of the INES, including an explanation of the various classification levels, can be found at <http://www.iaea.org/worldatom/inforesource/factsheets/ines.html>.

The NRC has participated in the INES program in a limited fashion since 1993. The NRC issued Generic Letter 92-09, "Limited Participation by NRC in the IAEA International Events Scale," dated December 12, 1992, to inform licensees that the agency had agreed to use the INES to rate all reactor events that result in the declaration of an "Alert" or higher emergency classification. Pursuant to that decision, from February 1993 through September 2001, the NRC transmitted a total of 32 reactor-related INES reports to the IAEA.

In order to be even more responsive to international stakeholders, the NRC has elected to increase its participation by evaluating all reported nuclear events (reactor, fuel cycle, materials, and transportation events) for possible rating on the INES. Medical misadministrations are outside the scope of the INES and will not be reviewed by the NRC for possible rating. Only events rated at Level 2 or higher will be reported to the IAEA, unless another member country specially requests the rating of a particular event. In 2002, there have been 2 INES ratings submitted to the IAEA [Davis Besse reactor vessel head degradation (INES 3) and the Point Beach potential common cause loss of auxiliary feedwater (INES 2)].

NEI's Presentation

Alan Nelson and Walt Lee provided the NEI presentation included in the attachment. This presentation provided an overview of INES ratings and industry concerns with INES application. NEI relayed its concerns that the on-site criteria for radiological barrier damage (fuel damage) are unclear or overly conservative. The concern is that there could be an overly conservative INES classification with respect to a relatively minor and expected fuel cladding failure. Industry representatives told the Staff that an incorrect characterization could result in damaging, unintended consequences such as a loss of public confidence and misinterpretation of a non-risk significant event.

NEI provided detailed comments on the INES User's Manual as provided in the attachments. These comments were primarily to include measurable quantitative criteria rather than the current INES qualitative criteria. NEI expressed that incorporation of these comments could enhance the consistency of reporting and the understanding of the ratings by industry since it provided measurable criteria.

NEI also discussed provisional ratings, which are INES ratings that are reported to the IAEA from preliminary information. NEI cautioned that reporting events with preliminary information may have unwarranted adverse impact on the utility. NEI preferred that all information provided to the public by both the NRC and the utility to be consistent and accurate. NEI emphasized that the INES reporting is a communication and public confidence issue. NEI requested that the utility have the opportunity to review and provide comments on the INES report before it is submitted to the IAEA to ensure accuracy and to prepare their own public relations response. This was especially important with events rated 2 or 3 that were reported to the IAEA that are not classified by any of the US emergency action levels. An INES rating of 2 or 3 may not warrant an emergency classification because these ratings characterize conditions adverse to quality or incidents rather than an accident.

The industry is preparing for the increased participation in the INES program. NEI desires to keep an ongoing dialogue on the INES program. They have asked the NRC staff to consider their written comments provided in the attachment. In 3-6 months, they would like to meet again to discuss the results of the NRC's review.

Terrence Reis

Terrence Reis, Section Chief
Operating Experience Section
Operating Reactor Improvements Program
Division of Regulatory Improvement Programs
Office of Nuclear Reactor Regulation

Enclosure: Attendees List

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/RA/
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Enclosure: Attendees List

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Attachment 1

# International Nuclear Event Scale (INES) Meeting

## July 26, 2002

Alan Nelson - NEI

Walt Lee - Southern Nuclear

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# Proposed Agenda

- Review purpose of the INES
- Seek to understand the process flowpath that will be used by NRC to classify and communicate an event classification using the INES wants to know our flowpath
- Review INES usage
- Discuss Industry concerns with INES application related to core damage events and fuel activity releases

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# INES Purpose

- Facilitates communication and understanding on the safety significance of events occurring at nuclear installations
- World wide audience includes:
  - The Nuclear Community
  - The Media
  - The Public



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# Process Flowpath?

- What is the process flowpath that is used by NRC to classify and communicate an event classification using the INES?
  - Is this a formalized process?
  - Does it include advance discussion with the affected utility, NEI, and/or INPO prior finalization of a classification?
  - Does it include advance notification to the affected utility, NEI, and/or INPO prior communication of the classification to IAEA?



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# INES Limitations

- "The scale does not replace the criteria already adopted nationally and inter-nationally for the technical analysis and reporting of events to safety authorities"
- "Nor does it form a part of the formal emergency arrangements that exist to deal with radiological accidents"

# INES Recommendation

- "Although broadly comparable, nuclear and radiological safety criteria and the terminology used to describe them vary from country to country"
- "The international scale has been designed to take account of this fact, but it is possible that user countries may wish to clarify the scale within their national context"



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# INES Usage

- Events are considered in terms of three different areas of impact:
  - Off-site impact
  - On-site impact
  - Defense-in-depth impact
- An event which has an impact on more than one area is always rated at the highest of the seven possible levels identified

# Off-Site Criteria Be

- Logical but dose assessment software might be challenged to create adequate comparisons
- Risk based given that it emulates public protection schemes utilized in current classification scheme
- If actually have a release then probably in a declared emergency
  - Proper INES classification will not be high priority
- Recommend adding British unit conversions

SI



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# On-Site Criteria

- On-Site criteria for radiological barrier damage (fuel damage) appears to be unclear or overly conservative
- INES classifies "severe core damage" at Level 5
  - Defined as more than a few % core inventory released from the fuel assemblies
    - IF assume that PWR (BWR) coolant activity would be 2e4 (1e3) uc/gm for a 100% gap activity release (source: RTM-96)
    - THEN PWR 3% core release ~ 600 uc/gm I-131 coolant activity
    - THEN BWR 3% core release ~ 30 uc/gm I-131 coolant activity
- INES classifies "significant core damage at Level 4
  - Defined as more than 0.1 % core inventory released from the fuel assemblies
    - THEN PWR 0.1% core release ~ 20 uc/gm I-131 coolant activity
    - THEN BWR 0.1% core release ~ 1 uc/gm I-131 coolant activity



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# Industry Concern

- Could result in an overly conservative INES classification of a relatively minor event
  - Could still be operating within Tech Spec limits
  - Risk informed Defense-in-Depth criteria overshadowed by On-site criteria

- Perception is reality
  - Incorrect characterization could result in damaging unintended consequences
    - Loss of public confidence
    - Misinterpretation of a non-risk significant event

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# On-Site Impact Level 5 Clarification

**Definition and Sheet 3 Note 1: Severe Damage to the reactor core or radiological barriers**
More than a few per cent of the fuel in a power reactor is molten or more than a few per cent of the core inventory has been released from the fuel assemblies. Incidents at other installations involving a major release of radioactivity on the site (comparable with the release from a core melt) with a serious off-site radiological safety threat. Examples of non-reactor accidents would be a major criticality accident, or a major fire or explosion releasing large quantities of activity within the installation.

**Recommended Change:**
More than 20 per cent of the fuel gap in a power reactor has been released into the reactor coolant and subsequently into the containment from the fuel assemblies. Incidents at other installations involving a major release of radioactivity on the site (comparable with a major release from the fuel clad gap) with a serious off-site radiological safety threat.

**Change Justification:**
A major release of radioactivity requiring offsite protective actions is not possible unless the containment barrier fails subsequent to a major failure of fuel cladding allowing radioactive material to be released from the core into the reactor coolant. 20 per cent fuel gap release is a value which indicates severe fuel damage. Regardless of whether containment is challenged, this amount of activity in containment, if released, could have such severe consequences that it is prudent to treat this as a potential loss of containment. NUREG-1228, "Source Estimations During Incident Response to Severe Nuclear Power Plant Accidents," indicates that such conditions do not exist when the amount of clad damage is less than 20%. This definition is consistent with the Emergency Action Level (EAL) classification methodology of NEI 99-01, Revision 4, for a General Emergency. Short-term, the evaluation of whether the activity release is a result of damaged clad due to fuel melting is irrelevant and would require either non-ALARA sampling/analysis and/or possible visual fuel inspection to determine.

# On-Site Impact Level 4 Clarification

**Definition and Sheet 3 Note 2: Significant damage to the reactor core or radiological barriers**
Any fuel melting has occurred or more than about 0.1% of the core inventory of a power reactor has been released from the fuel assemblies. Events at non-reactor installations involving the release of a few thousand terabecquerels of activity from their primary containment which cannot be returned to a satisfactory storage area.

**Recommended Change:**
More than a few per cent of the fuel gap (reactor coolant activity >300 μc/cc DEI) in a power reactor has been released into the reactor coolant and subsequently into the containment from the fuel assemblies. Events at non-reactor installations involving the release of a few thousand terabecquerels (8.1e4 Ci) of activity from their primary containment which cannot be returned to a satisfactory storage area.

**Change Justification:**
A release of radioactivity requiring on-site protective actions from core damage is not possible unless the containment barrier fails subsequent to a partial failure of fuel cladding allowing radioactive material to be released from the core into the reactor coolant. 5 per cent fuel gap release (reactor coolant activity >300 μc/cc DEI) is a concentration indicative of fuel damage several times larger than the maximum fuel leakage (including iodine spiking) allowed within technical specifications and is therefore indicative of significant fuel damage. This definition is consistent with the Emergency Action Level (EAL) classification methodology of NEI 99-01, Revision 4, for a Site Area Emergency. Escalation to level 5 would occur should activity levels rise to a 20% value. Short-term, the evaluation of whether the activity release is a result of damaged clad due to fuel melting is irrelevant and would require either non-ALARA sampling/analysis and/or possible visual fuel inspection to determine.

# On-Site Impact Level 3 Clarification

**Definition and Sheet 3 Note 3: Significant release from barriers which can be returned to a satisfactory storage area**
Events resulting in the release of a few thousand terabecquerels of activity into a secondary containment where the material can be returned to a satisfactory storage area.

**Recommended Change:**
More than a few per cent of the fuel gap (reactor coolant activity >300 μc/cc DEI) in a power reactor has been released into the reactor coolant from the fuel assemblies. Events resulting in a release of a few thousand terabecquerels (8.1e4 Ci) of activity into a secondary containment where the material can be returned to a satisfactory storage area.

**Change Justification:**
A release of radioactivity requiring on-site protective actions from core damage is not possible unless a partial failure of fuel cladding allows radioactive material to be released from the core into the reactor coolant. 5 per cent fuel gap release (reactor coolant activity >300 μc/cc DEI) is a concentration indicative of fuel damage several times larger than the maximum fuel leakage (including iodine spiking) allowed within technical specifications and is therefore indicative of fuel damage. With the fuel activity contained within the reactor coolant system, contamination spread may be controlled and activity levels may be reduced through installed isolation and cleanup systems. This definition is consistent with the Emergency Action Level (EAL) classification methodology of NEI 99-01, Revision 4, for an Alert Emergency. Escalation to level 4 would occur should significant reactor coolant leakage into containment subsequently occur.

# Defense in Depth Criteria

- Appears to be well developed and logical but somewhat difficult to use – time consuming
- Must use Initiator approach
  - Layer approach applies only to non-power reactors
- Has multiple examples provided for the user but excludes examples dealing with RCS activity and clad damage
  - Recommend examples be added for clarification

# Defense in Depth -Examples

- Level 2 Dose Equivalent Iodine all other isotopes excluding iodine
  - DEI or E-bar elevated out of normal operating limits requiring shutdown
- Level 1
  - DEI or E-bar elevated out of normal operating limits but returned to within normal operating limits within specified action statement time limits
- Level 0
  - DEI or E-bar elevated but within normal operating limits - no Mandatory LCO exists



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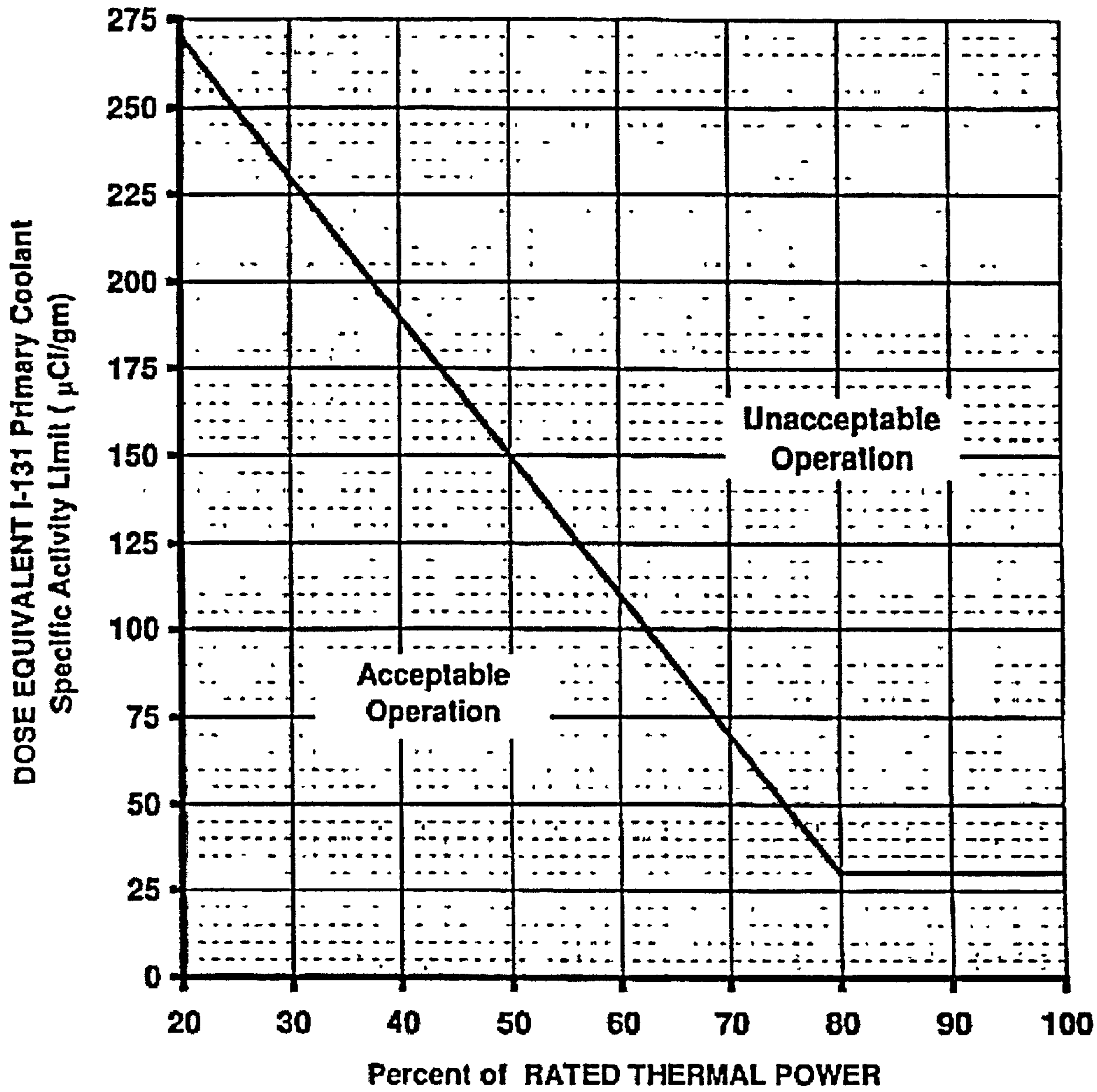
# PWR Technical Specification Example




25μC gm

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# Questions???



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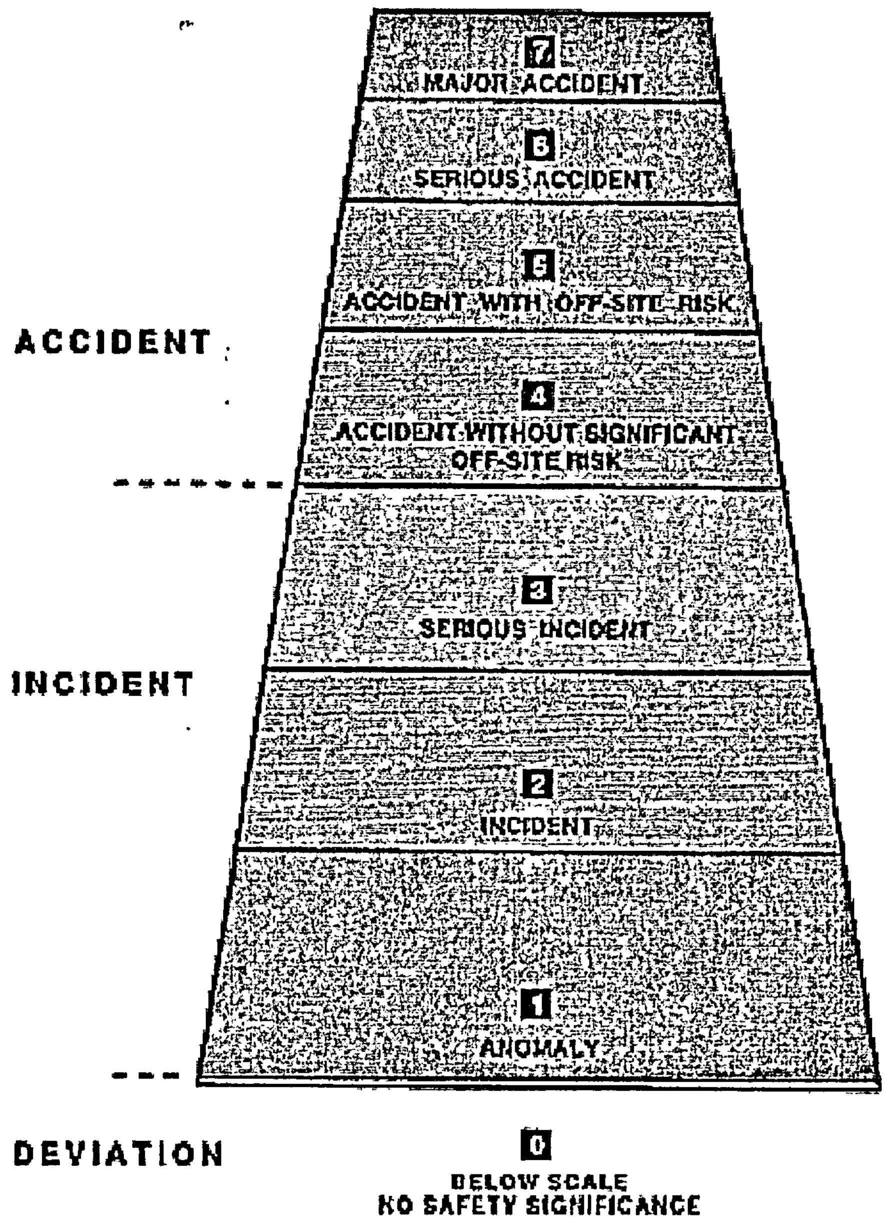
# The International Event Scale



| | AREA OF IMPACT | | |
|---|---|---|---|
| | OFF-SITE IMPACT | ON-SITE IMPACT | IMPACT ON DEFENCE IN DEPTH |
| 7<br>MAJOR ACCIDENT | MAJOR RELEASE: WIDESPREAD HEALTH AND ENVIRONMENTAL EFFECTS | | |
| 6<br>SERIOUS ACCIDENT | SIGNIFICANT RELEASE: LIKELY TO REQUIRE FULL IMPLEMENTATION OF PLANNED COUNTERMEASURES | | |
| 5<br>ACCIDENT WITH OFF-SITE RISK | LIMITED RELEASE: LIKELY TO REQUIRE PARTIAL IMPLEMENTATION OF PLANNED COUNTERMEASURES | SEVERE DAMAGE TO REACTOR CORE/RADIOLOGICAL BARRIERS | |
| 4<br>ACCIDENT WITHOUT SIGNIFICANT OFF-SITE RISK | MINOR RELEASE: PUBLIC EXPOSURE OF THE ORDER OF PRESCRIBED LIMITS | SIGNIFICANT DAMAGE TO REACTOR CORE/RADIOLOGICAL BARRIERS/FATAL EXPOSURE OF A WORKER | |
| 3<br>SERIOUS INCIDENT | VERY SMALL RELEASE: PUBLIC EXPOSURE AT A FRACTION OF PRESCRIBED LIMITS | SEVERE SPREAD OF CONTAMINATION/ACUTE HEALTH EFFECTS TO A WORKER | NEAR ACCIDENT NO SAFETY LAYERS REMAINING |
| 2<br>INCIDENT | | SIGNIFICANT SPREAD OF CONTAMINATION/ OVEREXPOSURE OF A WORKER | INCIDENTS WITH SIGNIFICANT FAILURES IN SAFETY PROVISIONS |
| 1<br>ANOMALY | | | ANOMALY BEYOND THE AUTHORIZED OPERATING REGIME |
| 0<br>DEVIATION | NO SAFETY SIGNIFICANCE | | |

Figure 1

Figure 2

| LEVEL/ DESCRIPTOR | NATURE OF THE EVENTS | EXAMPLES |
|---|---|---|
| 7 MAJOR ACCIDENT | • External release of a large fraction of the radioactive material in a large facility (e.g. the core of a power reactor). This would typically involve mixture of short and long lived radioactive fission products (in quantities radiologically equivalent to more than tens of thousands of terabecquerels of $^{131}I$). Such a release would result in the possibility of acute health effects; delayed health effects over a wide area, possibly involving more than one country; long term environmental consequences. | Chernobyl nuclear power plant, USSR (now in Ukraine), 1986 |
| 6 SERIOUS ACCIDENT | • External release of radioactive material (in quantities radiologically equivalent to the order of thousands to tens of thousands of terabecquerels of $^{131}I$). Such a release would be likely to result in full implementation of countermeasures covered by local emergency plans to limit serious health effects. | Kyshtym Reprocessing Plant, USSR (now in Russian Federation), 1957 |
| 5 ACCIDENT WITH OFF-SITE RISK | • External release of radioactive material (in quantities radiologically equivalent to the order of hundreds to thousands of terabecquerels of $^{131}I$). Such a release would be likely to result in partial implementation of countermeasures covered by emergency plans to lessen the likelihood of health effects.<br>• Severe damage to the installation. This may involve severe damage to a large fraction of the core of a power reactor, a major criticality accident or a major fire or explosion releasing large quantities of radioactivity within the installation. | Windscale Pile, UK, 1957<br>Three Mile Island nuclear power plant, USA, 1979 |
| 4 ACCIDENT WITHOUT SIGNIFICANT OFF-SITE RISK | • External release of radioactivity resulting in a dose to the critical group of the order of a few millisieverts.* With such a release the need for off-site protective actions would be generally unlikely except possibly for local food control.<br>• Significant damage to the installation. Such an accident might include damage leading to major on-site recovery problems such as partial core melt in a power reactor and comparable events at non-reactor installations.<br>• Irradiation of one or more workers resulting in an overexposure where a high probability of early death occurs. | Windscale Reprocessing Plant, UK, 1973<br>Saint Laurent nuclear power plant, France, 1980<br>Buenos Aires Critical Assembly, Argentina, 1983 |
| 3 SERIOUS INCIDENT | • External release of radioactivity resulting in a dose to the critical group of the order of tenths of millisieverts.* With such a release, off-site protective measures may not be needed.<br>• On-site events resulting in doses to workers sufficient to cause acute health effects and/or an event resulting in a severe spread of contamination for example a few thousand terabecquerels of activity released in a secondary containment where the material can be returned to a satisfactory storage area.<br>• Incidents in which a further failure of safety systems could lead to accident conditions, or a situation in which safety systems would be unable to prevent an accident if certain initiators were to occur | Vandellos nuclear power plant, Spain, 1989 |
| 2 INCIDENT | • Incidents with significant failure in safety provisions but with sufficient defence in depth remaining to cope with additional failures. These include events where the actual failures would be rated at level 1, but which reveal significant additional organizational inadequacies or safety culture deficiencies.<br>• An event resulting in a dose to a worker exceeding a statutory annual dose limit and/or an event which leads to the presence of significant quantities of radioactivity in the installation in areas not expected by design and which require corrective action. | |
| 1 ANOMALY | • Anomaly beyond the authorized regime, but with significant defence in depth remaining. This may be due to equipment failure, human error or procedural inadequacies and may occur in any area covered by the scale, e.g. plant operation, transport of radioactive material, fuel handling, and waste storage. Examples include breaches of technical specifications or transport regulations, incidents without direct safety consequences that reveal inadequacies in the organizational system or safety culture, minor defects in pipework beyond the expectations of the surveillance programme | |
| DEVIATION 0 | • Deviations where operational limits and conditions are not exceeded and which are properly managed in accordance with adequate procedures. Examples include a single random failure in a redundant system discovered during periodic inspections or tests, a planned reactor trip proceeding normally, spurious initiation of protection systems without significant consequences, leakages within the operational limits, minor spreads of contamination within controlled areas without wider implications for safety culture. | |



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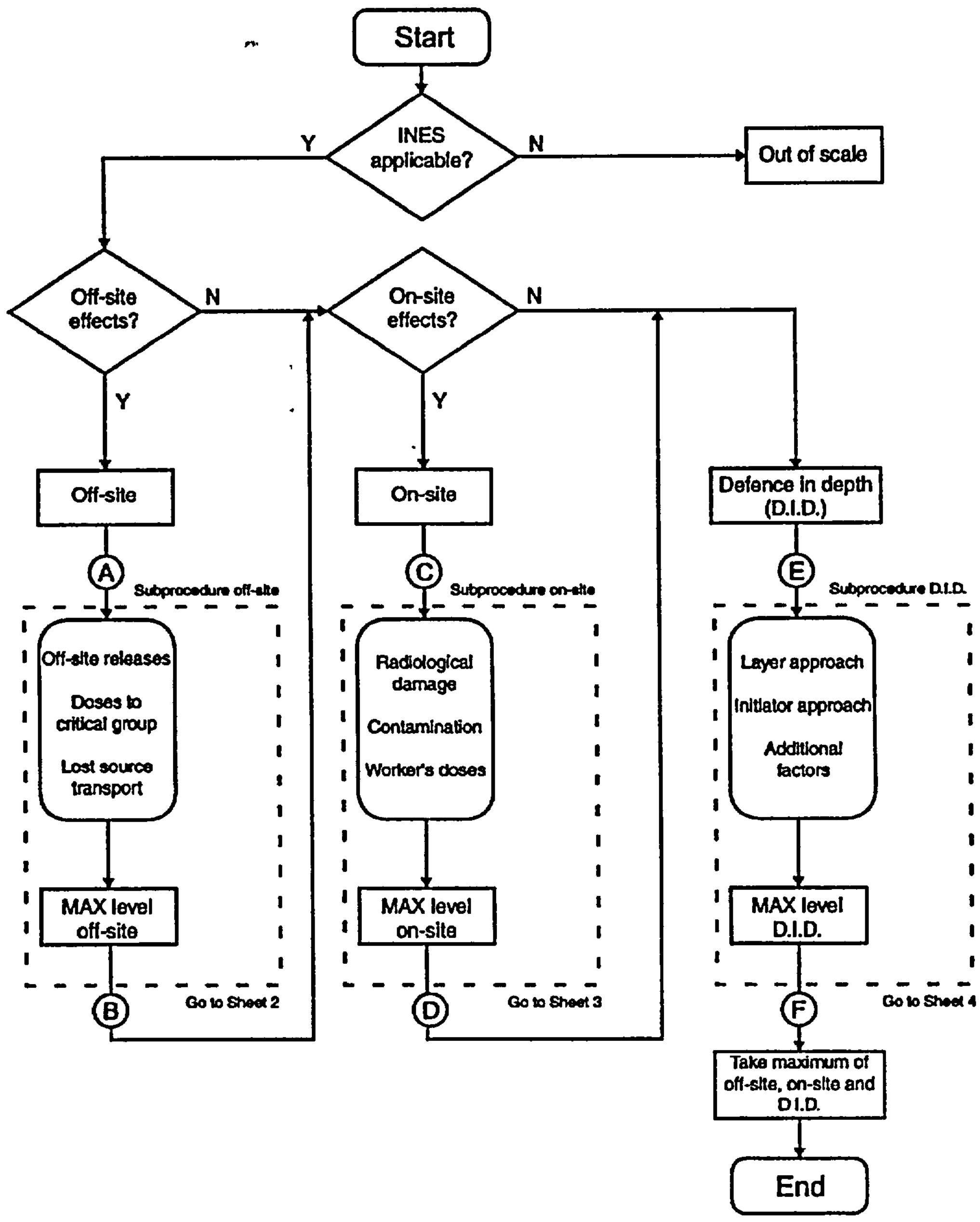
## Sheet 1
## INES rating procedures



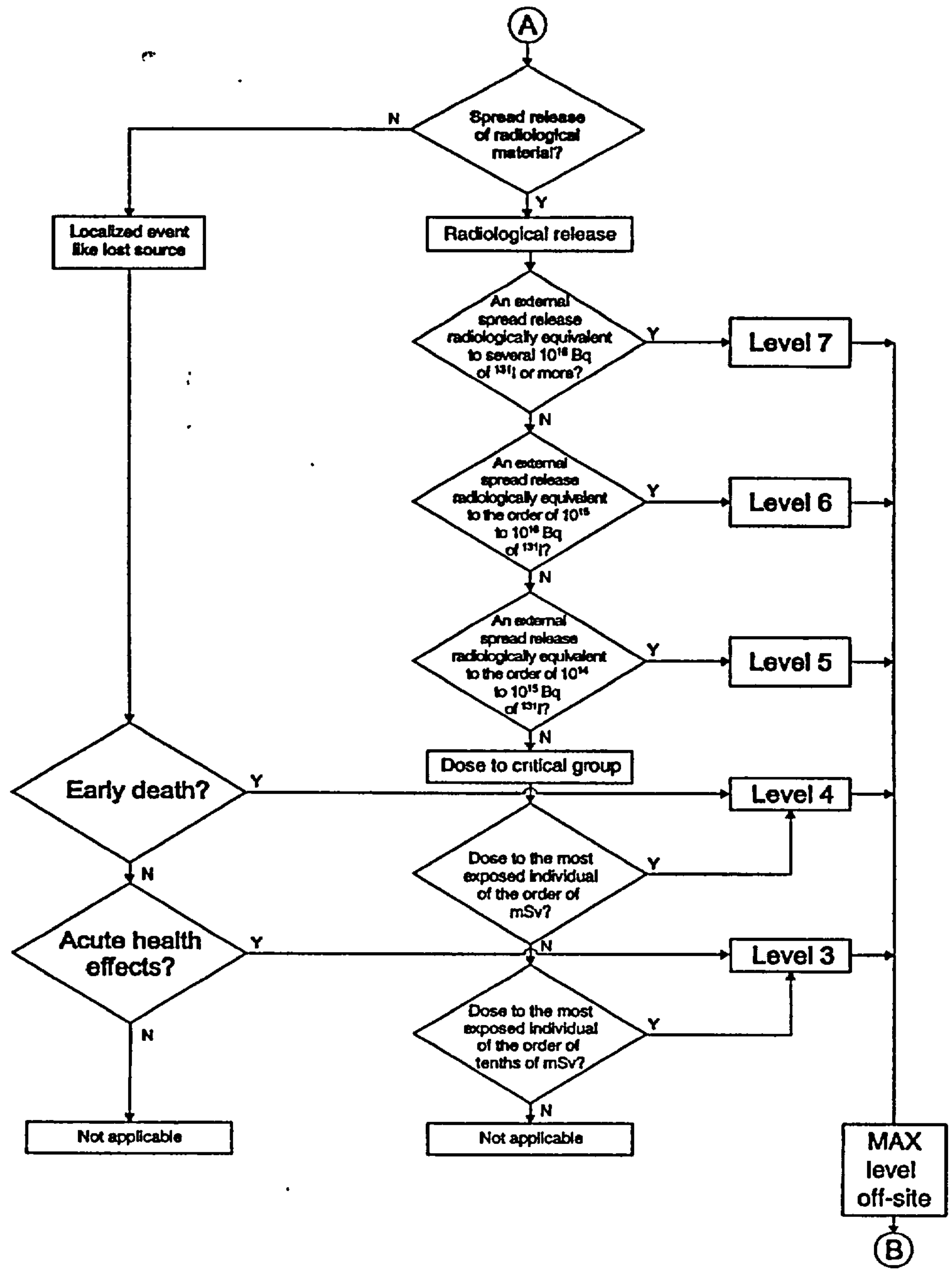
## Off-Site Impact

| | AREA OF IMPACT | | |
|---|---|---|---|
| | OFF-SITE IMPACT | ON-SITE IMPACT | IMPACT ON DEFENCE IN DEPTH |
| 7<br>MAJOR ACCIDENT | MAJOR RELEASE: WIDESPREAD HEALTH AND ENVIRONMENTAL EFFECTS | | |
| 6<br>SERIOUS ACCIDENT | SIGNIFICANT RELEASE: LIKELY TO REQUIRE FULL IMPLEMENTATION OF PLANNED COUNTERMEASURES | | |
| 5<br>ACCIDENT WITH OFF-SITE RISK | LIMITED RELEASE: LIKELY TO REQUIRE PARTIAL IMPLEMENTATION OF PLANNED COUNTERMEASURES | SEVERE DAMAGE TO REACTOR CORE/RADIOLOGICAL BARRIERS | |
| 4<br>ACCIDENT WITHOUT SIGNIFICANT OFF-SITE RISK | MINOR RELEASE: PUBLIC EXPOSURE OF THE ORDER OF PRESCRIBED LIMITS | SIGNIFICANT DAMAGE TO REACTOR CORE/RADIOLOGICAL BARRIERS/FATAL EXPOSURE OF A WORKER | |
| 3<br>SERIOUS INCIDENT | VERY SMALL RELEASE: PUBLIC EXPOSURE AT A FRACTION OF PRESCRIBED LIMITS | SEVERE SPREAD OF CONTAMINATION/ACUTE HEALTH EFFECTS TO A WORKER | NEAR ACCIDENT NO SAFETY LAYERS REMAINING |
| 2<br>INCIDENT | | SIGNIFICANT SPREAD OF CONTAMINATION/ OVEREXPOSURE OF A WORKER | INCIDENTS WITH SIGNIFICANT FAILURES IN SAFETY PROVISIONS |
| 1<br>ANOMALY | | | ANOMALY BEYOND THE AUTHORIZED OPERATING REGIME |
| 0<br>DEVIATION | NO SAFETY SIGNIFICANCE | | |

*FIG. 1. Basic structure of the scale (the criteria given in the matrix are broad indicators only).*

[redacted]

Sheet 2
Subprocedure off-site



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# Off-Site Impact Definitions

*Level 7. Major release* - Definition: **An external release corresponding to a quantity of radioactivity radiologically equivalent to a release to the atmosphere of several tens of thousands of terabecquerels of 131 I or more (>2.7e5 curies I-131).**

This corresponds to the release of a large fraction of the core inventory of a power reactor, typically involving a mixture of short and long lived radioactive fission products. With such a release, there is a possibility of acute health effects. Delayed health effects over a wide area, perhaps involving more than one country, are expected. Long term environmental consequences are also likely.

*Level 6. Significant release* - Definition: **An external release corresponding to a quantity of radioactivity radiologically equivalent to a release to the atmosphere of the order of thousands to tens of thousands of terabecquerels of 131 I (2.7e4 to 2.7e5 curies I-131).**

With such a release it is very likely that protective measures such as sheltering and evacuation will be judged to be necessary to limit health effects on members of the public over the emergency planning zone.

*Level 5. Limited release* - Definition: **An external release, corresponding to a quantity of radioactivity radiologically equivalent to a release to the atmosphere of the order of hundreds to thousands of terabecquerels of 131 I (2.7e3 to 2.7e4 curies I-131).**

As a result of the actual release, some protective measures will probably be required, for example, localized sheltering and/or evacuation to minimize the likelihood of health effects.

*Level 4. Minor release* - Definition: **An external release of radioactivity resulting in a dose (as defined in Section III–1.3) to the critical group of the order of a few millisieverts (300 mr) or an event, such as a lost source or transport event, which results in a dose to a member of the public of greater than 5 Gy (500 rad) (i.e. one with a high probability of early death).**

As a result of the actual release, off-site protective actions are generally unlikely, except for possible local food controls. Other actions can nevertheless be taken as a precaution against further degradation of the plant's status. Plant status is taken into account in the other areas of impact (on-site impact and impact on defence in depth).

*Level 3. Very small release* - Definition: **An external release of radioactivity resulting in a dose (as defined in Section III–1.3) to the critical group of the order of tenths of a millisievert (10mr) or an event, such as a lost source or transport event, which results in a dose to a member of the public leading to acute health effects (such as whole body exposure of the order of 1 Gy (100 rad) and body surface exposure of the order of 10 Gy (1000 rad).**

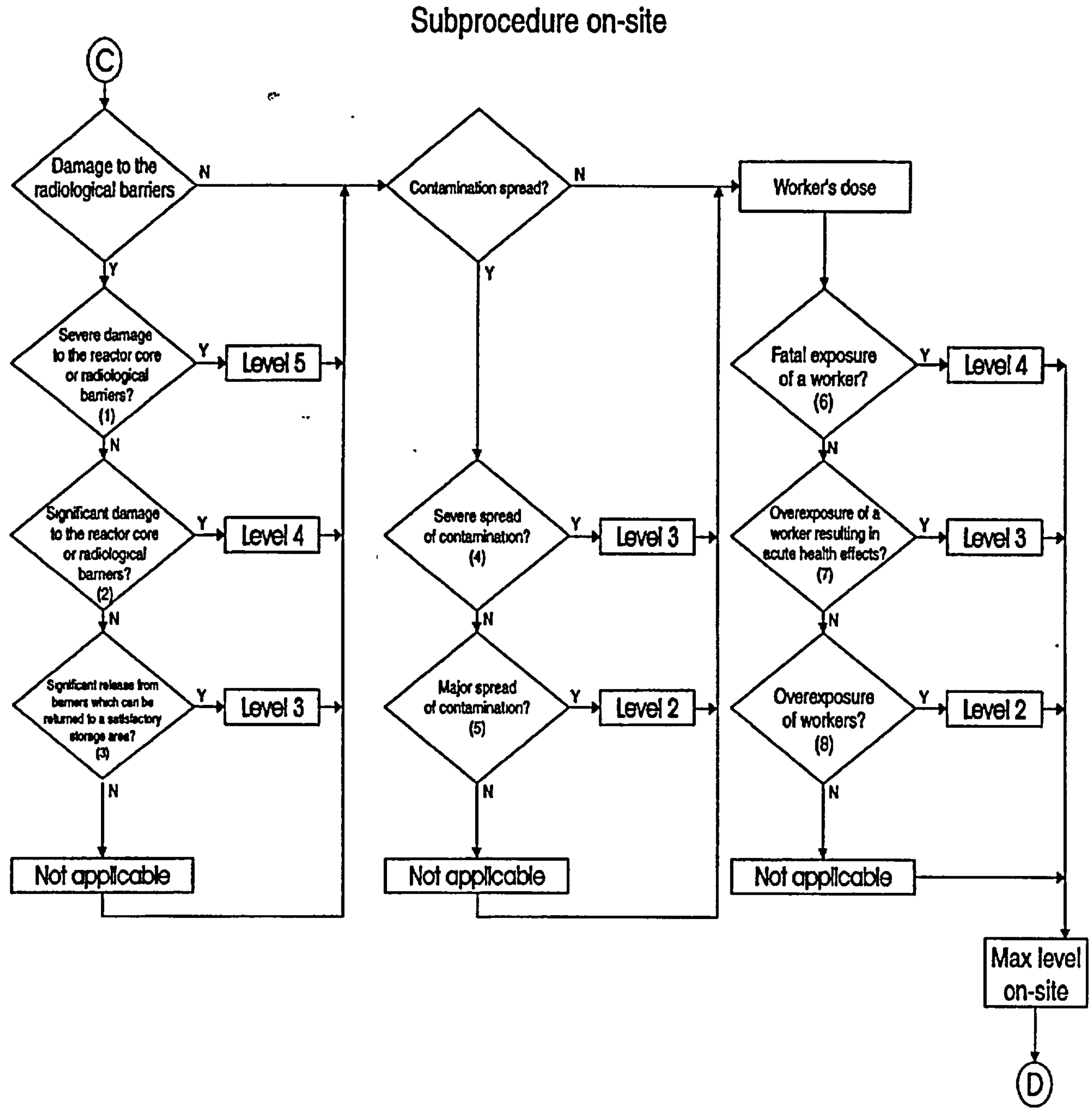
Following such an actual release, off-site protection measures are not needed. Such measures can nevertheless be taken as a precaution against further degradation of the plant's status. Plant status is taken into account in the other areas of impact (on-site impact and impact on defence in depth).

On-Site Impact

| | AREA OF IMPACT | | |
|---|---|---|---|
| | OFF-SITE IMPACT | ON-SITE IMPACT | IMPACT ON DEFENCE IN DEPTH |
| 7 MAJOR ACCIDENT | MAJOR RELEASE: WIDESPREAD HEALTH AND ENVIRONMENTAL EFFECTS | | |
| 6 SERIOUS ACCIDENT | SIGNIFICANT RELEASE: LIKELY TO REQUIRE FULL IMPLEMENTATION OF PLANNED COUNTERMEASURES | | |
| 5 ACCIDENT WITH OFF-SITE RISK | LIMITED RELEASE: LIKELY TO REQUIRE PARTIAL IMPLEMENTATION OF PLANNED COUNTERMEASURES | SEVERE DAMAGE TO REACTOR CORE/RADIOLOGICAL BARRIERS | |
| 4 ACCIDENT WITHOUT SIGNIFICANT OFF-SITE RISK | MINOR RELEASE: PUBLIC EXPOSURE OF THE ORDER OF PRESCRIBED LIMITS | SIGNIFICANT DAMAGE TO REACTOR CORE/RADIOLOGICAL BARRIERS/FATAL EXPOSURE OF A WORKER | |
| 3 SERIOUS INCIDENT | VERY SMALL RELEASE: PUBLIC EXPOSURE AT A FRACTION OF PRESCRIBED LIMITS | SEVERE SPREAD OF CONTAMINATION/ACUTE HEALTH EFFECTS TO A WORKER | NEAR ACCIDENT NO SAFETY LAYERS REMAINING |
| 2 INCIDENT | | SIGNIFICANT SPREAD OF CONTAMINATION/ OVEREXPOSURE OF A WORKER | INCIDENTS WITH SIGNIFICANT FAILURES IN SAFETY PROVISIONS |
| 1 ANOMALY | | | ANOMALY BEYOND THE AUTHORIZED OPERATING REGIME |
| 0 DEVIATION | NO SAFETY SIGNIFICANCE | | |

*FIG. 1. Basic structure of the scale (the criteria given in the matrix are broad indicators only).*

## Sheet 3
## Subprocedure on-site



# On-Site Impact Level 4 Contamination

**Definition and Sheet 3 Note 4: Severe spread of contamination**
Events resulting in a contamination level which did or easily could have resulted in one or more workers receiving a dose leading to acute health effects (such as whole body exposures of the order of 1 Gy (100 rad) and body surface exposures of the order of 10 Gy (1000 rad).

# On-Site Impact Level 3 Contamination

**Definition and Sheet 3 Note 5: Major spread of contamination**
Events leading to the presence of significant quantities of radioactivity in the installation, in areas not expected by design and which require corrective action. In this context 'significant quantity' should be interpreted as:

(a) Contamination by liquids involving a total activity radiologically equivalent to a few hundred gigabecquerels (8.1 Ci) of 106 Ru

(b) A spillage of solid radioactive material of radiological significance equivalent to the order of a few hundred gigabecquerels (8.1 Ci) of 106 Ru, providing the surface and airborne contamination levels exceed ten times those permitted for operating areas

(c) A release of airborne radioactive material, contained within a building and involving quantities of radiological significance equivalent to the order of a few tens of gigabecquerels (0.81 Ci) of 131 I.

## On-Site Impact Level 4 Worker Dose

**Definition and Sheet 3 Note 6: Fatal Exposure of a Worker**
External irradiation of one or more workers, which results in an overexposure where a high probability of early death occurs (about 5 Gy) (500 rad).

## On-Site Impact Level 3 Worker Dose

**Definition and Sheet 3 Note 7: Overexposure of a worker resulting in acute health effects**
Events resulting in a dose rate or contamination level which resulted in one or more workers receiving a dose leading to acute health effects (such as whole body exposures of the order of 1 Gy (100rad) and body surface exposures of the order of 10 Gy (1000 rad).

## On-Site Impact Level 2 Worker Dose

**Definition and Sheet 3 Note 8: Overexposure of workers**
An event resulting in a dose to one or more workers exceeding an International Commission for Radiological Protection annual dose limit for radiation workers. An event resulting in the need for significant surgery to prevent a dose that would otherwise have been about an order of magnitude above the annual dose limit.


NEI

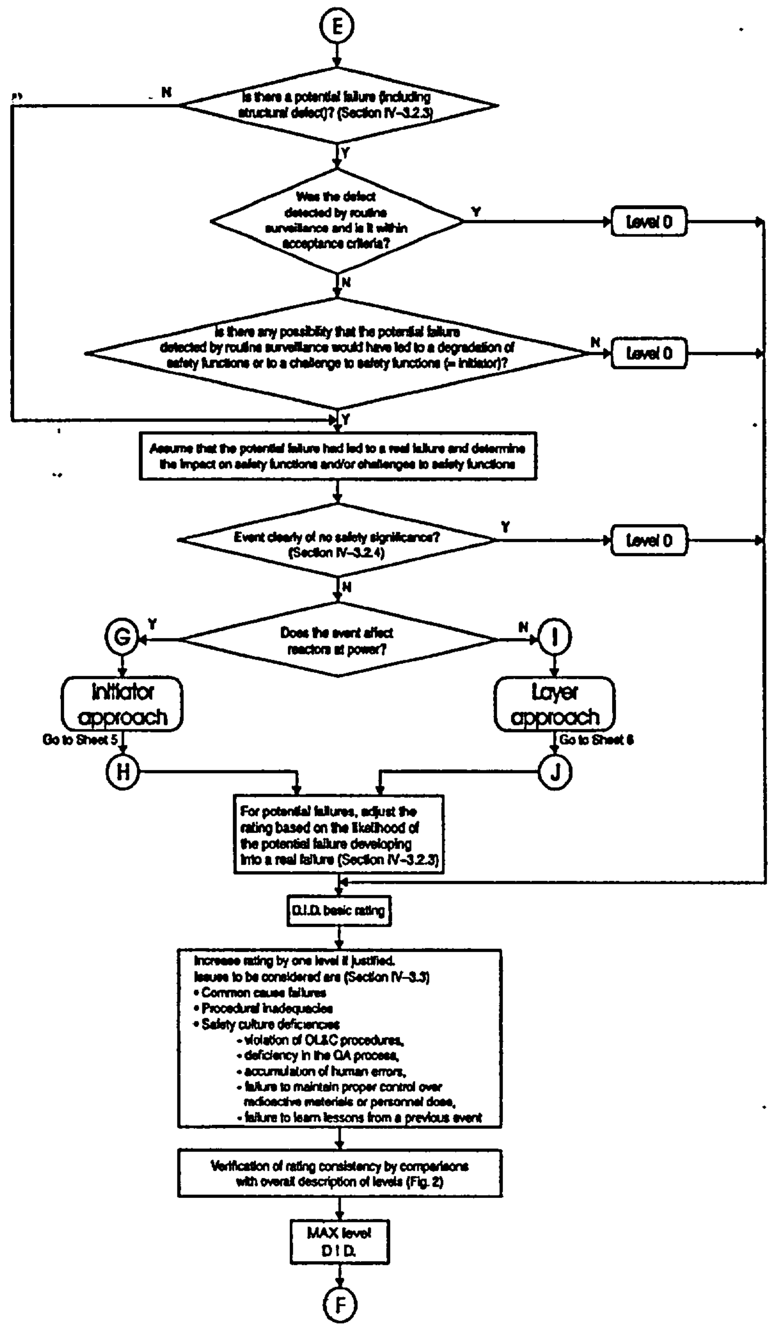
Figure 1

| | AREA OF IMPACT | | |
|---|---|---|---|
| | OFF-SITE IMPACT | ON-SITE IMPACT | IMPACT ON DEFENCE IN DEPTH |
| 7<br>MAJOR ACCIDENT | MAJOR RELEASE: WIDESPREAD HEALTH AND ENVIRONMENTAL EFFECTS | | |
| 6<br>SERIOUS ACCIDENT | SIGNIFICANT RELEASE: LIKELY TO REQUIRE FULL IMPLEMENTATION OF PLANNED COUNTERMEASURES | | |
| 5<br>ACCIDENT WITH OFF-SITE RISK | LIMITED RELEASE: LIKELY TO REQUIRE PARTIAL IMPLEMENTATION OF PLANNED COUNTERMEASURES | SEVERE DAMAGE TO REACTOR CORE/RADIOLOGICAL BARRIERS | |
| 4<br>ACCIDENT WITHOUT SIGNIFICANT OFF-SITE RISK | MINOR RELEASE: PUBLIC EXPOSURE OF THE ORDER OF PRESCRIBED LIMITS | SIGNIFICANT DAMAGE TO REACTOR CORE/RADIOLOGICAL BARRIERS/FATAL EXPOSURE OF A WORKER | |
| 3<br>SERIOUS INCIDENT | VERY SMALL RELEASE: PUBLIC EXPOSURE AT A FRACTION OF PRESCRIBED LIMITS | SEVERE SPREAD OF CONTAMINATION/ACUTE HEALTH EFFECTS TO A WORKER | NEAR ACCIDENT NO SAFETY LAYERS REMAINING |
| 2<br>INCIDENT | | SIGNIFICANT SPREAD OF CONTAMINATION/ OVEREXPOSURE OF A WORKER | INCIDENTS WITH SIGNIFICANT FAILURES IN SAFETY PROVISIONS |
| 1<br>ANOMALY | | | ANOMALY BEYOND THE AUTHORIZED OPERATING REGIME |
| 0<br>DEVIATION | NO SAFETY SIGNIFICANCE | | |

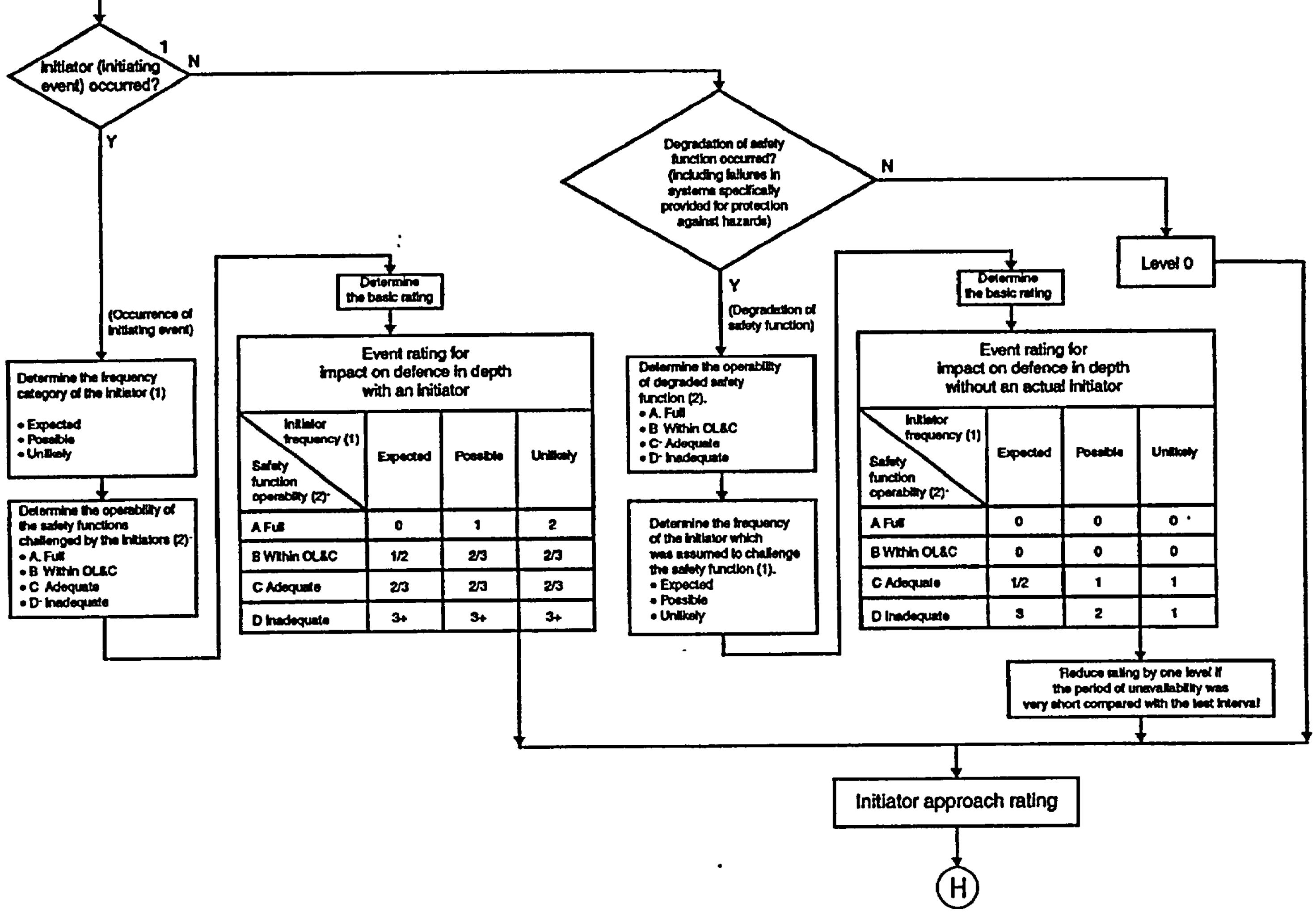
*FIG. 1. Basic structure of the scale (the criteria given in the matrix are broad indicators only).*

NEI

## Sheet 4
## Subprocedure defence in depth (D.I.D.)



## Sheet 5
## Initiator approach




Event rating for impact on defence in depth with an initiator

| Safety function operability (2) \ Initiator frequency (1) | Expected | Possible | Unlikely |
|---|---|---|---|
| A Full | 0 | 1 | 2 |
| B Within OL&C | 1/2 | 2/3 | 2/3 |
| C Adequate | 2/3 | 2/3 | 2/3 |
| D Inadequate | 3+ | 3+ | 3+ |

Event rating for impact on defence in depth without an actual initiator

| Safety function operability (2) \ Initiator frequency (1) | Expected | Possible | Unlikely |
|---|---|---|---|
| A Full | 0 | 0 | 0 |
| B Within OL&C | 0 | 0 | 0 |
| C Adequate | 1/2 | 1 | 1 |
| D Inadequate | 3 | 2 | 1 |

# NEI Review of NRC Participation in theInternational Nuclear Event Scale

The primary purpose of the INES is to facilitate communication and understanding between the nuclear community, the media, and the public on the safety significance of events occurring at nuclear installations. In the INES, events are considered in terms of three different areas of impact: off-site impact, on-site impact and impact on defense in depth. An event which has an impact on more than one area is always rated at the highest level identified.

The NRC has modified their participation in the International Nuclear Event Scale (INES) as of January 14, 2002. This change is in response to increasing interest by foreign government agencies and media to events occurring at facilities in the United States. This includes reactor, fuel cycle, materials and transportation but excludes medical misadministration. Licensees should continue to report events in accordance to regulations. The INES is not intended to supersede the existing four-tier emergency classification system.

NRC has participated in the INES in a "limited" manner since 1993, sending a total of 32 reactor-related INES reports to the IAEA. It now plans to report all nuclear events, including reactor, fuel cycle, materials, and transportation events.

In the past the NRC notified IAEA of US licensee events of Alert or higher. Last year (2001) U.S. licensees declared 4 Alerts - Fire (2), leakage, and toxic gas.

On the INES scale this relates to a 4-7 termed "accidents." Level 4 is an "accident without significant off-site risk;" minor release: public exposure of the order of prescribed limits; significant damage to reactor core/radiological barriers/fatal exposure to worker; no impact on defense in-depth.

NRC plans, however, to submit only events rated at Level 2 or higher. Levels 1 and 0 are considered, respectively, an "anomaly" and non-safety significant; Levels 4-7 are the most serious. NRC estimates the change will result in about one reactor report and fewer than five materials reports each year.

The INES classifies "severe core damage" as a level 5 accident. It classifies "significant core damage" as a level 4 accident. There is no category that includes minor fuel or core damage. There is a concern that a minor fuel damage event could be misrepresented or misclassified on the INES as a more serious (level 4) event than in actuality.

The attached draft revision to the International Nuclear Event Scale (INES) Users Manual provides recommended changes to clarify the document during events which involve fuel activity releases. Currently the "on-site impact wording" could result in an inappropriate INES level being assigned.

The INES Users Manual states that:

> "The scale does not replace the criteria already adopted nationally and inter-nationally for the technical analysis and reporting of events to safety authorities. Nor does it form a part of the formal emergency arrangements that exist in each country to deal with radiological accidents." "Although broadly comparable, nuclear and radiological safety criteria and the terminology used to describe them vary from country to country. The international scale has been designed to take account of this fact, but it is possible that user countries may wish to clarify the scale within their national context."

It is within the above context that the following recommended changes are submitted. Revised text from the INES Users Manual excerpt (pages 2-36) is either highlighted in yellow or blue. A summary table of key changes is also attached (pages 37-38). Reference material including British conversion units and a copy of a plant specific RCS activity Technical Specification follow (pages 39-44).

## Examples of INES Ratings

Level/Example: 7-Chernobyl, USSR (1986), 5- TMI-2 (1979), 4-Tokai-mura, Japan (1999), 3-Davis Besse (2002), 2-Texas overexposure (2002) 0-Indian Point (2000)



Attachment 2

**DRAFT**

# THE INTERNATIONAL NUCLEAR EVENT SCALE (INES) USER'S MANUAL

## 2001 EDITION

Jointly prepared by IAEA and OECD/NEA

**DRAFT**

# Part I

# SUMMARY DESCRIPTION

## I–1. INTRODUCTION

### I–1.1. Background

The International Nuclear Event Scale (INES) is a means for promptly communicating to the public in consistent terms the safety significance of events reported at nuclear installations. By putting events into proper perspective, it can facilitate common understanding among the nuclear community, the media and the public.

The scale was designed by an international group of experts convened jointly in 1989 by the IAEA and the Nuclear Energy Agency of the Organisation for Economic Co-operation and Development (OECD/NEA). It also reflects the experience gained from the use of similar scales in France and Japan as well as from consideration of possible scales in several other countries.

Initially the scale was applied for a trial period to classify events at nuclear power plants, and then extended and adapted to enable it to be applied to all installations associated with the civil nuclear industry. It is now operating successfully in over 60 countries. This edition of the INES User's Manual can be applied to any event associated with radioactive material and/or radiation and to any event occurring during the transport of radioactive material.

### I-1.2 General description of the scale

"Events are classified on the scale at seven levels: the upper levels (4–7) are termed "accidents" and the lower levels (1–3) "incidents". Events which have no safety significance are classified below scale at level 0 and are termed "deviations". Events which have no safety relevance are termed "out of scale". The structure of the scale is shown in Fig. 1, in the form of a matrix with key words. The words used are not intended to be precise or definitive. Each level is defined in detail in Parts III and IV of this manual. Events are considered in terms of three different areas of impact represented by each of the columns: off-site impact, on-site impact and impact on defence in depth.

The first column relates to events resulting in off-site releases of radioactivity. Since this is the only possible direct impact on the public, such releases are under-standably of particular concern. Thus, the lowest point in this column represents a release giving the critical group an estimated radiation dose numerically equivalent to about one-tenth of the annual dose limit for the public; this is classified as level 3. Such a dose is also typically about one-tenth of the average annual dose received from natural background radiation. The highest level is a major nuclear accident with widespread health and environmental consequences.

The second column considers the on-site impact of the event. This category covers a range from level 2 (contamination and/or overexposure of a worker) to level 5 (severe damage to the reactor core or radiological barriers).

All nuclear facilities are designed and operated so that a succession of safety layers act to prevent major off-site or on-site impact and the extent of the safety layers provided generally will be commensurate with the potential for such impacts. These safety layers must all fail before substantial off-site or on-site consequences occur. The provision of these layers is termed "defence in depth". The third column relates to incidents in which these defence in depth provisions have been degraded. This column spans the incident levels from 1 to 3.

An event which has an impact on more than one area is always rated at the highest level identified. Events which do not reach the threshold in any of the three areas are rated below scale at level 0."

### I–1.3. Scope of the scale

The scale can be applied to any event associated with radioactive material and/or radiation and to any event occurring during the transport of radioactive material. It does not classify industrial accidents or other events which are not related to nuclear or radiological operations. Such events are termed "out of scale". For example, although events associated with a turbine or generator can affect safety related equipment, faults affecting only the availability of a turbine or generator would be classified as out of scale. Similarly, events such as fires would be classified as out of scale if they did not involve any possible radiological hazard and did not affect the safety layers.

The scale does not apply to those controls provided only for the safeguarding of fissile material. Equally, published accountancy imbalances for fissile material (material unaccounted for (MUF)) would be classified as out of scale.

**I-1.4 Using the scale**

Although broadly comparable, nuclear and radiological safety criteria and the terminology used to describe them vary from country to country. The international scale has been designed to take account of this fact, but it is possible that user countries may wish to clarify the scale within their national context.

The detailed rating procedures are provided in this manual. The INES leaflet should not be used as the basis for rating events as it only provides examples of events at each level, rather than actual definitions.

The scale is designed for prompt use following an event. However, there will be occasions when a longer time-scale is required to understand and rate the consequences of an event. In these rare circumstances, a provisional rating will be given with confirmation at a later date. It is also possible that as a result of further information, an event may require re-rating."

Although the scale is used for all facilities, it is physically impossible at some types of installation for events to occur which involve the release to the environment of considerable quantities of radioactive material. For these installations, the upper levels of the scale would not be applicable. These include research reactors, unirradiated nuclear fuel treatment facilities and waste storage sites.

The scale does not replace the criteria already adopted nationally and inter-nationally for the technical analysis and reporting of events to safety authorities. Nor does it form a part of the formal emergency arrangements that exist in each country to deal with radiological accidents.

The scale is not appropriate as the basis for selecting events for feedback of operational experience, as important lessons can often be learnt from events of relatively minor significance.

Finally, it is not appropriate to use this scale to compare safety performance between countries. Each country has different arrangements for reporting minor events to the public, and it is difficult to ensure precise international consistency in rating events at the boundary between level 0 and level 1. Although information will be available generally on events at level 2 and above on the scale, the statistically small number of such events, which also varies from year to year, makes it difficult to provide meaningful international comparisons.

**I–1.5. Examples of rated nuclear events**

**I–1.6. Structure of the manual**

This manual consists of six parts:

—Part I provides an overview of the scale,
—Part II is a summary of the procedure to be used to rate events and to report them to the INES information service,
—Part III gives the detailed guidance required to rate events in terms of off-site and on-site impact,
—Part IV provides the detailed guidance required to rate events in terms of their impact on defence in depth,
—Part V consists of examples to illustrate the use of the rating guidance,
—Part VI contains a number of appendices giving detailed information on particular aspects of the scale.

# Part II

# RATING PROCEDURE AND REPORTING EVENTS TO THE IAEA

## II–1. RATING PROCEDURE

The flow chart provided on the following pages briefly describes the INES rating procedure for rating any event associated with radioactive material and/or radiation and any event occurring during the transport of radioactive material. The format of the flow chart is intended to show the logical route to be followed to assess the safety significance of any event. It provides an overview for those new to rating events and a summary of the procedure for those familiar with the INES User's Manual. It cannot, of course, be used in isolation from the detailed guidance provided in Parts III and IV. The computer software INESAR (INES Automatic Rating) has been developed on the basis of a similar earlier flow chart.

## II–2. COMMUNICATING EVENTS TO THE IAEA INFORMATION SERVICE

The INES National Officer is committed to communicate as quickly as possible (target: within 24 hours) official information on the consequences of an event to all the participating countries (see Appendix VI) through the IAEA INES Information Service. The criteria for identifying which events should be communicated are:

(a) Events rated at level 2 and above,
(b) Events attracting international public interest.

The information is presented in a specific format using the 'Event Rating Form' available from the IAEA. This form is forwarded to the IAEA INES Information Service through two redundant channels, fax machine and electronic mail. The INES Information Service is always in operation and can therefore ensure dissemination of the form at any time.

Draft Excerpt

| | AREA OF IMPACT | | |
| --- | --- | --- | --- |
| | OFF-SITE IMPACT | ON-SITE IMPACT | IMPACT ON DEFENCE IN DEPTH |
| 7<br>MAJOR ACCIDENT | MAJOR RELEASE: WIDESPREAD HEALTH AND ENVIRONMENTAL EFFECTS | | |
| 6<br>SERIOUS ACCIDENT | SIGNIFICANT RELEASE: LIKELY TO REQUIRE FULL IMPLEMENTATION OF PLANNED COUNTERMEASURES | | |
| 5<br>ACCIDENT WITH OFF-SITE RISK | LIMITED RELEASE: LIKELY TO REQUIRE PARTIAL IMPLEMENTATION OF PLANNED COUNTERMEASURES | SEVERE DAMAGE TO REACTOR CORE/RADIOLOGICAL BARRIERS | |
| 4<br>ACCIDENT WITHOUT SIGNIFICANT OFF-SITE RISK | MINOR RELEASE: PUBLIC EXPOSURE OF THE ORDER OF PRESCRIBED LIMITS | SIGNIFICANT DAMAGE TO REACTOR CORE/RADIOLOGICAL BARRIERS/FATAL EXPOSURE OF A WORKER | |
| 3<br>SERIOUS INCIDENT | VERY SMALL RELEASE: PUBLIC EXPOSURE AT A FRACTION OF PRESCRIBED LIMITS | SEVERE SPREAD OF CONTAMINATION/ACUTE HEALTH EFFECTS TO A WORKER | NEAR ACCIDENT NO SAFETY LAYERS REMAINING |
| 2<br>INCIDENT | | SIGNIFICANT SPREAD OF CONTAMINATION/ OVEREXPOSURE OF A WORKER | INCIDENTS WITH SIGNIFICANT FAILURES IN SAFETY PROVISIONS |
| 1<br>ANOMALY | | | ANOMALY BEYOND THE AUTHORIZED OPERATING REGIME |
| 0<br>DEVIATION | NO SAFETY SIGNIFICANCE | | |

*FIG. 1. Basic structure of the scale (the criteria given in the matrix are broad indicators only).*

| LEVEL/ DESCRIPTOR | NATURE OF THE EVENTS | EXAMPLES |
|---|---|---|
| 7 MAJOR ACCIDENT | • External release of a large fraction of the radioactive material in a large facility (e g the core of a power reactor) This would typically involve mixture of short and long lived radioactive fission products (in quantities radiologically equivalent to more than tens of thousands of terabecquerels of $^{131}I$). Such a release would result in the possibility of acute health effects; delayed health effects over a wide area, possibly involving more than one country; long term environmental consequences. | Chernobyl nuclear power plant, USSR (now in Ukraine), 1986 |
| 6 SERIOUS ACCIDENT | • External release of radioactive material (in quantities radiologically equivalent to the order of thousands to tens of thousands of terabecquerels of $^{131}I$). Such a release would be likely to result in full implementation of countermeasures covered by local emergency plans to limit serious health effects. | Kyshtym Reprocessing Plant, USSR (now in Russian Federation), 1957 |
| 5 ACCIDENT WITH OFF-SITE RISK | • External release of radioactive material (in quantities radiologically equivalent to the order of hundreds to thousands of terabecquerels of $^{131}I$). Such a release would be likely to result in partial implementation of countermeasures covered by emergency plans to lessen the likelihood of health effects.<br>• Severe damage to the installation. This may involve severe damage to a large fraction of the core of a power reactor, a major criticality accident or a major fire or explosion releasing large quantities of radioactivity within the installation. | Windscale Pile, UK, 1957<br>Three Mile Island nuclear power plant, USA, 1979 |
| 4 ACCIDENT WITHOUT SIGNIFICANT OFF-SITE RISK | • External release of radioactivity resulting in a dose to the critical group of the order of a few millisieverts.* With such a release the need for off-site protective actions would be generally unlikely except possibly for local food control.<br>• Significant damage to the installation. Such an accident might include damage leading to major on-site recovery problems such as partial core melt in a power reactor and comparable events at non-reactor installations.<br>• Irradiation of one or more workers resulting in an overexposure where a high probability of early death occurs. | Windscale Reprocessing Plant, UK, 1973<br>Saint Laurent nuclear power plant, France, 1980<br>Buenos Aires Critical Assembly, Argentina, 1983 |
| 3 SERIOUS INCIDENT | • External release of radioactivity resulting in a dose to the critical group of the order of tenths of millisieverts.* With such a release, off-site protective measures may not be needed.<br>• On-site events resulting in doses to workers sufficient to cause acute health effects and/or an event resulting in a severe spread of contamination for example a few thousand terabecquerels of activity released in a secondary containment where the material can be returned to a satisfactory storage area.<br>• Incidents in which a further failure of safety systems could lead to accident conditions, or a situation in which safety systems would be unable to prevent an accident if certain initiators were to occur. | Vandellos nuclear power plant, Spain, 1989 |
| 2 INCIDENT | • Incidents with significant failure in safety provisions but with sufficient defence in depth remaining to cope with additional failures These include events where the actual failures would be rated at level 1, but which reveal significant additional organizational inadequacies or safety culture deficiencies.<br>• An event resulting in a dose to a worker exceeding a statutory annual dose limit and/or an event which leads to the presence of significant quantities of radioactivity in the installation in areas not expected by design and which require corrective action. | |
| 1 ANOMALY | • Anomaly beyond the authorized regime, but with significant defence in depth remaining This may be due to equipment failure, human error or procedural inadequacies and may occur in any area covered by the scale, e g plant operation, transport of radioactive material, fuel handling, and waste storage Examples include breaches of technical specifications or transport regulations, incidents without direct safety consequences that reveal inadequacies in the organizational system or safety culture, minor defects in pipework beyond the expectations of the surveillance programme | |
| DEVIATION 0 | • Deviations where operational limits and conditions are not exceeded and which are properly managed in accordance with adequate procedures. Examples include, a single random failure in a redundant system discovered during periodic inspections or tests, a planned reactor trip proceeding normally, spurious initiation of protection systems without significant consequences, leakages within the operational limits, minor spreads of contamination within controlled areas without wider implications for safety culture. | |

* The doses are expressed in terms of effective dose equivalent (whole dose body) Those criteria, where appropriate, can also be expressed in terms of corresponding annual effluent discharge limits authorized by national authorities.

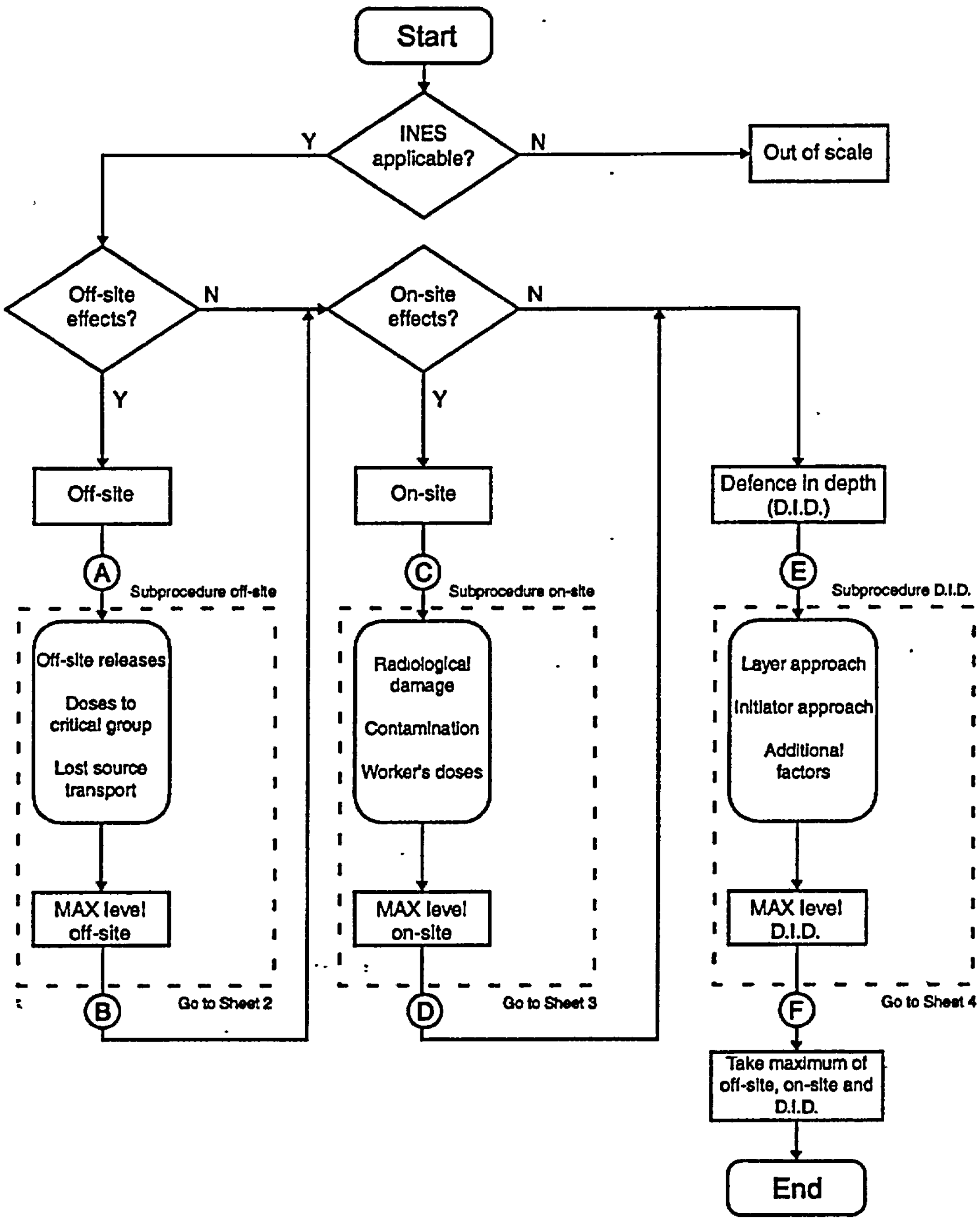
5 - Large fraction needs to be defined to clearly scope "severe". Words here are not consistent with section 3 definitions. See section 3 for suggested changes.

4 - "Partial Core Melt" needs to be deleted and replaced with "significant fuel clad activity release". See section 3 for suggested changes.

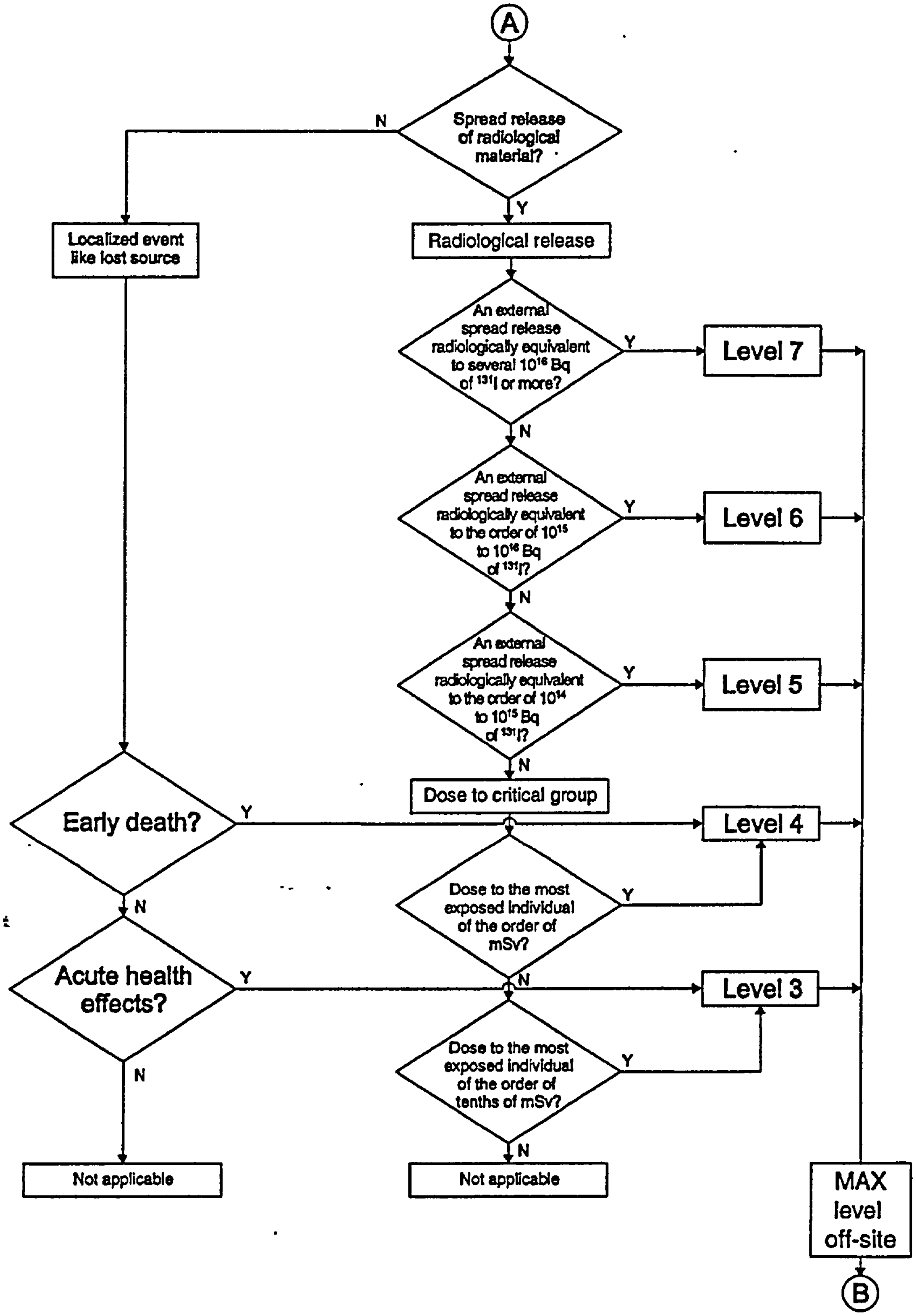
3 - "Severe spread of contamination" example needs to be further defined. See section 3 for suggested changes.

*FIG. 2. The International Nuclear Event Scale (for prompt communication of safety significance).*

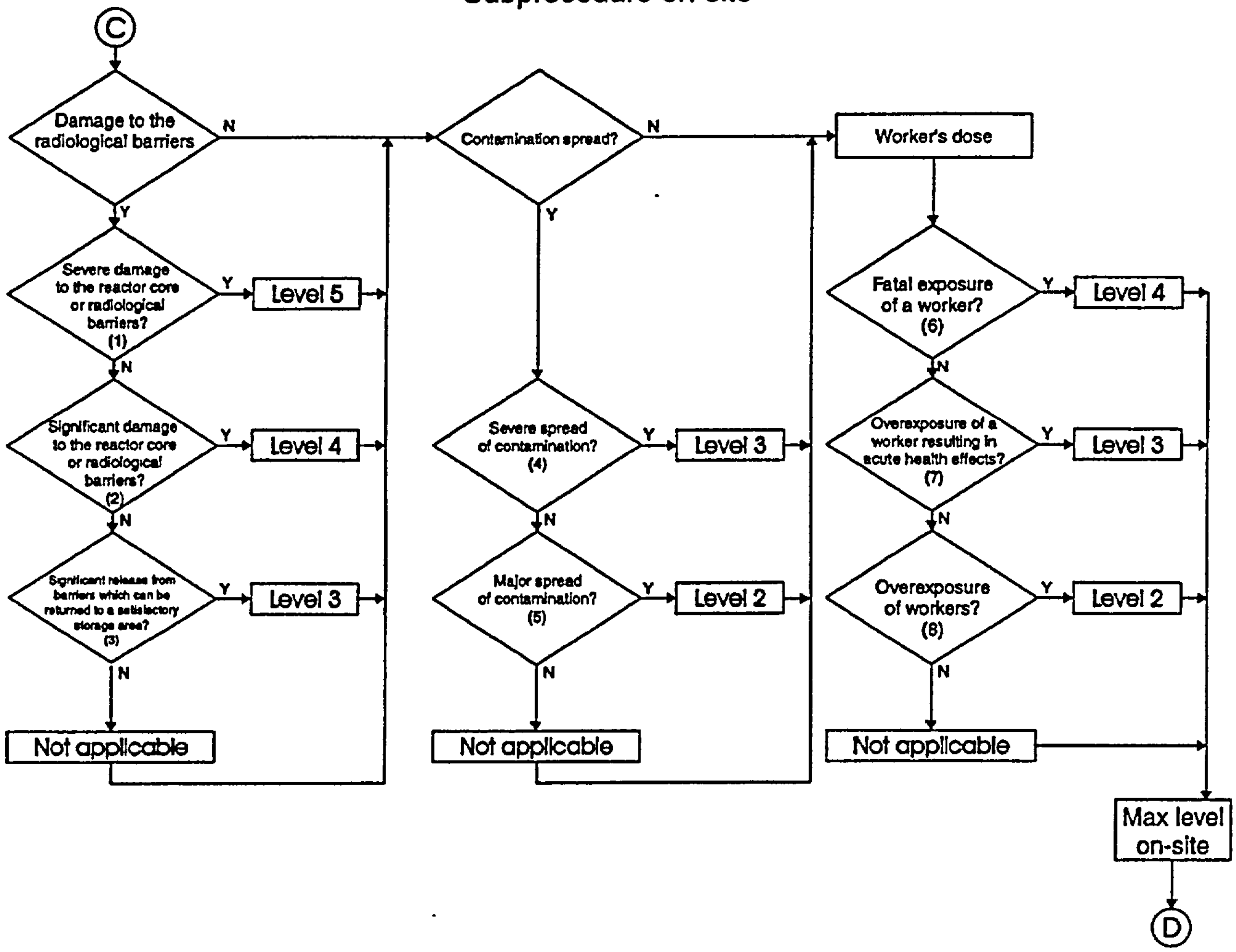
## Sheet 1
## INES rating procedures



Sheet 2
Subprocedure off-site



## Sheet 3
## Subprocedure on-site



## Notes for Sheet 3:

1. More than a few per cent of the fuel in a power reactor is molten or more than a few per cent of the core inventory has been released from the fuel assemblies. Incidents at other installations involving a major release of radioactivity on the site (comparable with the release from core melt) with a serious off-site radiological safety threat.

   Recommended Change:

   1. More than 20 per cent of the fuel gap in a power reactor has been released into the reactor coolant and subsequently into the containment from the fuel assemblies. Incidents at other installations involving a major release of radioactivity on the site (comparable with a major release from the fuel clad gap) with a serious off-site radiological safety threat.

   Change Justification:

   A major release of radioactivity requiring offsite protective actions is not possible unless the containment barrier fails subsequent to a major failure of fuel cladding allowing radioactive material to be released from the core into the reactor coolant. 20 per cent fuel gap release is a value which indicates severe fuel damage. Regardless of whether containment is challenged, this amount of activity in containment, if released, could have such severe consequences that it is prudent to treat this as a potential loss of containment. NUREG-1228, "Source Estimations During Incident Response to Severe Nuclear Power Plant Accidents," indicates that such conditions do not exist when the amount of clad damage is less than 20%. This definition is consistent with the Emergency Action Level (EAL) classification methodology of NEI 99-01, Revision 4, for a General Emergency. Short-term, the evaluation of whether the activity release is a result of damaged clad due to fuel melting is irrelevant and would require either non-ALARA sampling/analysis and/or possible visual fuel inspection to determine.

2. Any fuel melting has occurred or more than about 0.1% of the core inventory of a power reactor has been released from the fuel assemblies. Events at non-reactor installations involving the release of a few thousand terabecquerels of activity from their primary containment which cannot be returned to a satisfactory storage area.

   Recommended Change:

   2. More than a few per cent of the fuel gap (reactor coolant activity >300 μc/cc DEI) in a power reactor has been released into the reactor coolant and subsequently into the containment from the fuel assemblies. Events at non-reactor installations involving the release of a few thousand terabecquerels (8.1e4 Ci) of activity from their primary containment which cannot be returned to a satisfactory storage area.

   Change Justification:

   A release of radioactivity requiring on-site protective actions from core damage is not possible unless the containment barrier fails subsequent to a partial failure of fuel cladding allowing radioactive material to be released from the core into the reactor coolant. 5 per cent fuel gap release (reactor coolant activity >300 μc/cc DEI) is a concentration indicative of fuel damage several times larger than the maximum fuel leakage (including iodine spiking) allowed within technical specifications and is therefore indicative of significant fuel damage. This definition is consistent with the Emergency Action Level (EAL) classification methodology of NEI 99-01, Revision 4, for a Site Area Emergency. Escalation to level 5 would occur should activity levels rise to a 20% value. Short-term, the evaluation of whether the activity release is a result of damaged clad due to fuel melting is irrelevant and would require either non-ALARA sampling/analysis and/or possible visual fuel inspection to determine.

3. Events resulting in a release of a few thousand terabecquerels of activity into a secondary containment where the material can be returned to a satisfactory storage area.

   Recommended Change:

   3. More than a few per cent of the fuel gap (reactor coolant activity >300 μc/cc DEI) in a power reactor has been released into the reactor coolant from the fuel assemblies. Events resulting in a release of a few thousand terabecquerels (8.1e4 Ci) of activity into a secondary containment where the material can be returned to a satisfactory storage area.

   Change Justification:

   A release of radioactivity requiring on-site protective actions from core damage is not possible unless a partial failure of fuel cladding allows radioactive material to be released from the core into the reactor coolant. 5 per cent fuel gap release (reactor coolant activity >300 μc/cc DEI) is a concentration indicative of fuel damage several times larger than the maximum fuel leakage (including iodine spiking) allowed within technical specifications and is therefore indicative of fuel damage. With the fuel activity contained within the reactor coolant system, contamination spread may be controlled and activity levels may be reduced through installed isolation and cleanup systems. This definition is consistent with the Emergency Action Level (EAL) classification methodology of NEI 99-01, Revision 4, for an Alert Emergency. Escalation to level 4 would occur should significant reactor coolant leakage into containment subsequently occur.

4. Events resulting in a dose rate or contamination level which could easily have resulted in one or more workers receiving a dose leading to acute health effects (such as whole body exposure of the order of 1 Gy (100 rad) and body surface exposures of the order of 10 Gy (1000 rad)).

5. An event resulting in the sum of gamma plus neutron dose rates of greater than 50 mSv per hour (5000 mr per hour) in a plant operating area (dose rate measured 1 m from the source). An event leading to the presence of significant quantities of radioactivity in the installation, inareas not expected by design (see Section III–2.3) and which requires corrective action. In this

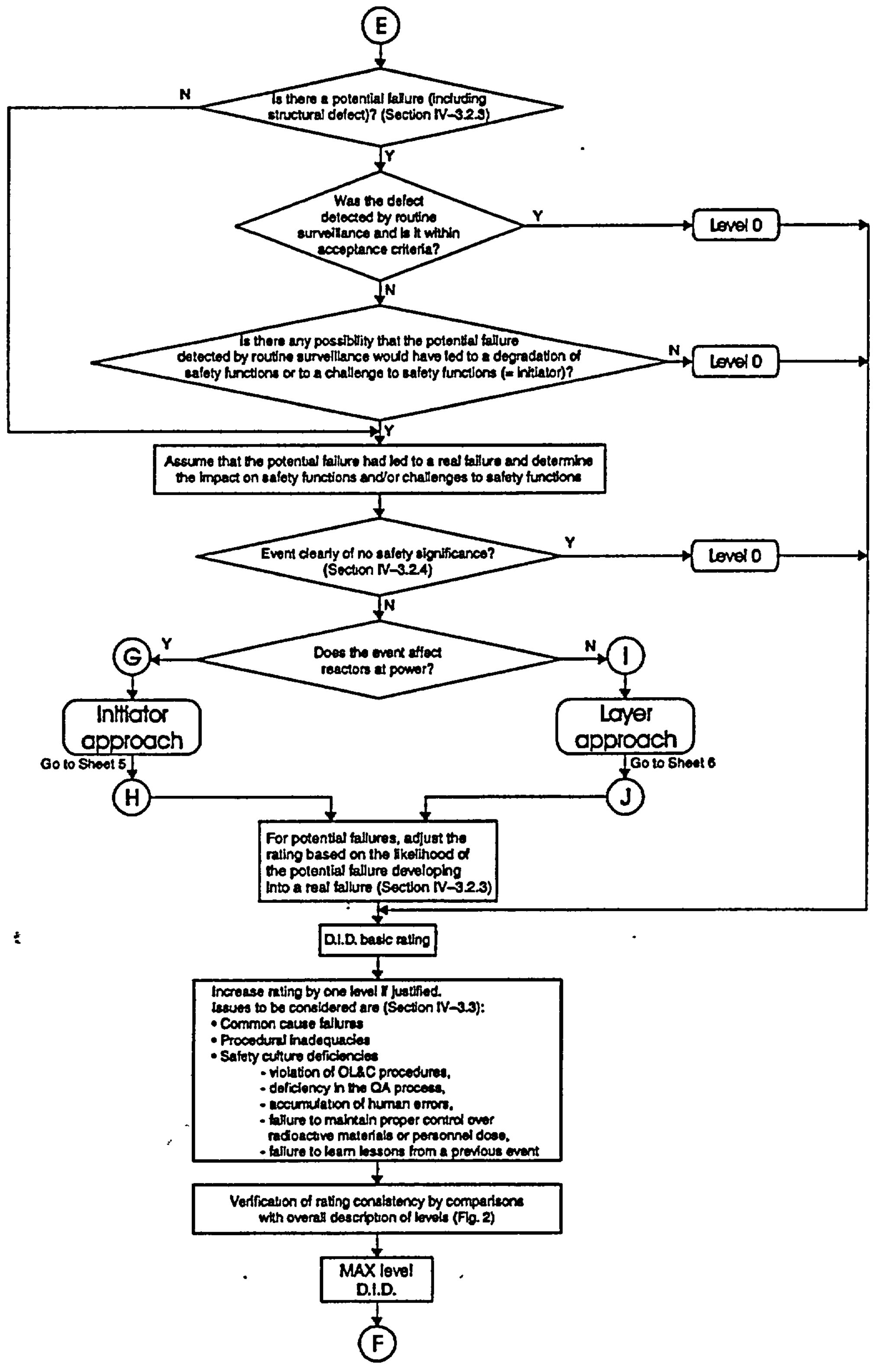
context, "significant quantity" should be interpreted as: (a) contamination by liquids involving a total activity radiologically equivalent to a few hundred giga-becquerels of 106 Ru; (b) a spillage of solid radioactive material of radiological significance equivalent to the order of a few hundred gigabecquerels of 106 Ru, providing the surface and airborne contamination levels exceed ten times those permitted for con-trolled areas; (c) a release of airborne radioactive material, contained within a building and involving quantities of radiological significance equivalent to the order of a few tens of gigabecquerels of 131 I.

6. External irradiation of one or more workers, which results in an overexposure where a high probability of early death occurs (about 5 Gy) (500 rad).

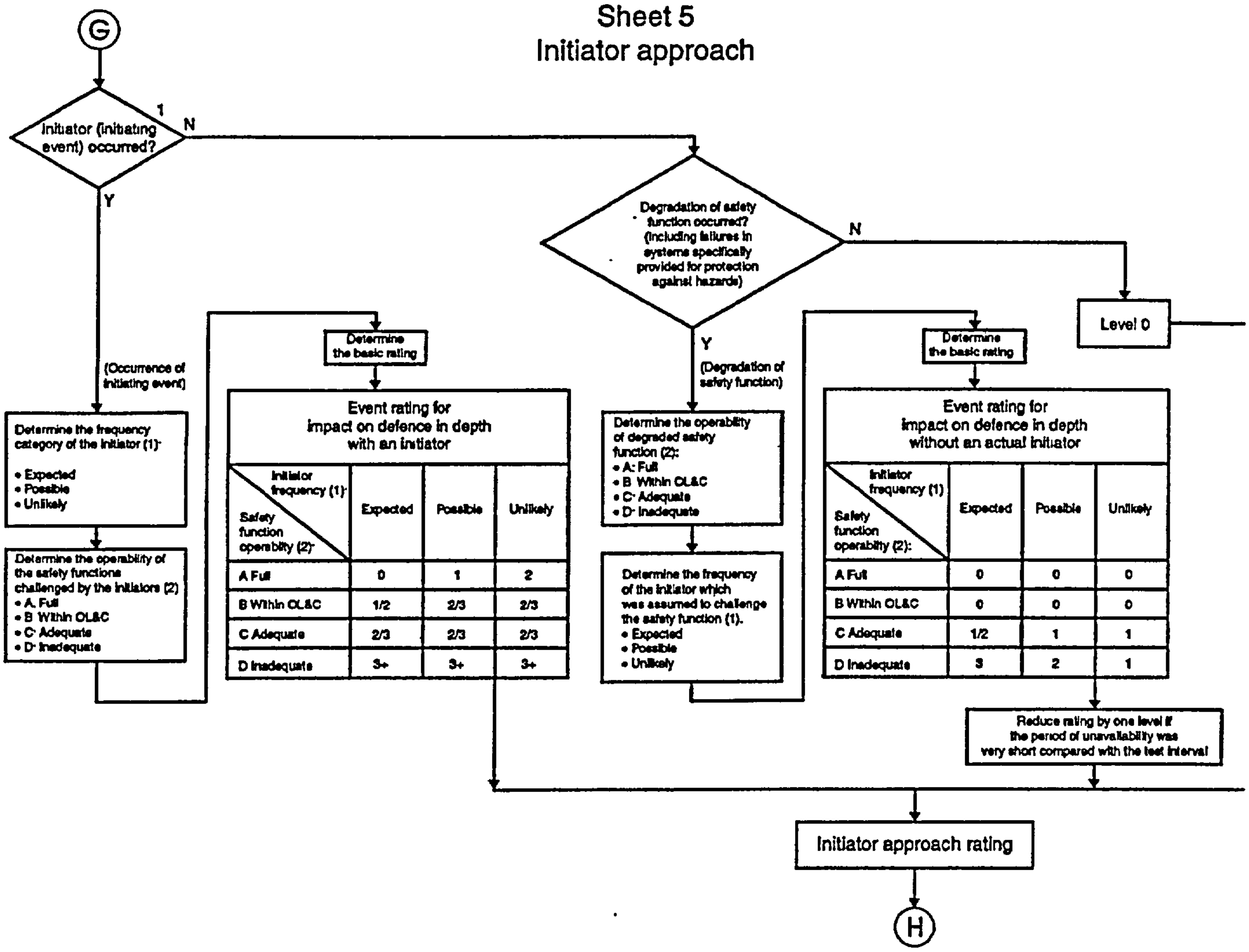
7. Events resulting in a dose rate or contamination level which resulted in one or more workers receiving a dose leading to acute health effects (such as whole body exposures of the order of 1 Gy (100rad) and body surface exposures of the order of 10 Gy (1000 rad)).

8. An event resulting in a dose to one or more workers exceeding an International Commission for Radiological Protection annual dose limit for radiation workers. An event resulting in the need for significant surgery to prevent a dose that would otherwise have been about an order of magnitude above the annual dose limit.

## Sheet 4
## Subprocedure defence in depth (D.I.D.)



## Sheet 5
## Initiator approach




**Event rating for impact on defence in depth with an initiator**

| Safety function operability (2) \ Initiator frequency (1) | Expected | Possible | Unlikely |
|---|---|---|---|
| A Full | 0 | 1 | 2 |
| B Within OL&C | 1/2 | 2/3 | 2/3 |
| C Adequate | 2/3 | 2/3 | 2/3 |
| D Inadequate | 3+ | 3+ | 3+ |

**Event rating for impact on defence in depth without an actual initiator**

| Safety function operability (2): \ Initiator frequency (1) | Expected | Possible | Unlikely |
|---|---|---|---|
| A Full | 0 | 0 | 0 |
| B Within OL&C | 0 | 0 | 0 |
| C Adequate | 1/2 | 1 | 1 |
| D Inadequate | 3 | 2 | 1 |

## Notes for Sheet 5

1. Definition of initiator and initiator frequency: An initiator is an occurrence that challenges the safety systems and requires them to function. In practice, the initiator may be different from the occurrence which starts the event. Frequency categories of the initiators are as follows:

- Expected: initiators which are expected to occur once or several times during the life of the plant.
- Possible: initiators which are not 'expected', but have an anticipated frequency during the plant lifetime of greater than about 1% (i.e. about $3 \times 10^{-4}$/a).
- Unlikely: initiators considered in the design of the plant which are less likely than the above.

2. Safety function operability: The three basic safety functions are: (a) controlling the reactivity or the process conditions; (b) cooling the radioactive material; (c) confining the radioactive material. The function is achieved by safety systems, including support systems such as electrical supplies, cooling and instrument supplies. To provide a framework for rating events, four levels of operability are considered:

A — Full: all safety systems and components provided by the design to cope with the particular initiator are fully operable.
B — Minimum required (by operational limits and conditions (OL&C)): minimum operability of safety systems specified in the OL&C for continued operation at power, even for a limited time.
C — Adequate: a level of operability of safety systems sufficient to achieve the particular safety function for the initiator being considered.
D — Inadequate: the degraded operability of the safety systems is such that the safety function cannot be fulfilled.

# Part III

# OFF-SITE AND ON-SITE IMPACT

## III-1. OFF-SITE MPACT

### III-1.1 General Description

The rating of events in terms of the off-site impact takes account of the actual radiological impact outside the site of the nuclear installation. This can be expressed in terms of the amount of activity released from a facility or the assessed dose to members of the public. It is accepted that for a significant accident at a facility, it will not be possible to determine with accuracy at an early stage the size of the off-site release. However, it should be possible to indicate the release in broad terms and thus to assign the accident to a tentative level on the scale. It is possible that subsequent re-evaluation of the extent of the release would necessitate revision of the initial estimate of the rating of the event on the scale.

It is important to note that the extent of emergency response to accidents is not used as a basis for rating. Details of the planning against accidents at nuclear plants vary from one country to another and it is also possible that precautionary measures may be taken in some cases even where they are not fully justified by the actual size of the release. For these reasons, it is the size of the release and the assessed dose which should be used to rate the event on the scale and not the protective actions taken in response to emergency plans.

Five levels have been selected, starting from level 7, where a large fraction of the core inventory of a commercial nuclear power plant is released, down to level 3, where the dose to a member of the public is numerically equivalent to about one tenth of the annual dose limit. For levels 3 and 4, the committed dose to the critical group is used to assess the appropriate level. For levels 5–7, the definitions are in terms of a quantity of activity released, radiologically equivalent to a given number of terabecquerels of 131 I. The reason for the change is that for these larger releases the actual dose received will depend very much on the countermeasures implemented.

The release levels were set on the basis that, taking account of the likely countermeasures, it was estimated that a level 5 release could give doses of the order of ten times the doses defined for level 4. Of course, the actual quantity of radioactivity release corresponding to the threshold for level 5 is significantly more than an order of magnitude greater than the minimum release size that would correspond to a level 4 accident.

Below level 3, off-site impact is considered as being insignificant for the purpose of rating an event on the scale. Only the on-site impact and the impact on defence in depth have to be considered at these lower levels.

Events considered under off-site impact will be of two types, both of which are considered in the definition given below. The first relates to releases that will be dispersed significantly so that the doses will be small but to a significant number of members of the public. The second refers to doses, such as could occur from a lost source or a transport event, that may be larger but to a much smaller number of people. Specific guidance is given for this latter type of event in the definitions for levels 3 and 4. The definitions of levels 5–7 apply to both types of events.

### III-1.2 Definition of levels

*Level 7. Major release*

Definition: **An external release corresponding to a quantity of radioactivity radiologically equivalent to a release to the atmosphere of several tens of thousands of terabecquerels of 131 I or more (>2.7e5 curies I-131).**

This corresponds to the release of a large fraction of the core inventory of a power reactor, typically involving a mixture of short and long lived radioactive fission products. With such a release, there is a possibility of acute health effects. Delayed health effects over a wide area, perhaps involving more than one country, are expected. Long term environmental consequences are also likely.

*Level 6. Significant release*

Definition: **An external release corresponding to a quantity of radioactivity radiologically equivalent (see footnote 1) to a release to the atmosphere of the order of thousands to tens of thousands of terabecquerels of 131 I (2.7e4 to 2.7e5 curies I-131).**

With such a release it is very likely that protective measures such as sheltering and evacuation will be judged to be necessary to limit health effects on members of the public over the emergency planning zone.

*Level 5. Limited release*

Definition: **An external release, corresponding to a quantity of radioactivity radiologically equivalent (see note 1) to a release to the atmosphere of the order of hundreds to thousands of terabecquerels of 131 I (2.7e3 to 2.7e4 curies I-131).**

As a result of the actual release, some protective measures will probably be required, for example, localized sheltering and/or evacuation to minimize the likelihood of health effects.

*Level 4. Minor release*

Definition: **An external release of radioactivity resulting in a dose (as defined in Section III–1.3) to the critical group of the order of a few millisieverts (300 mr) or an event, such as a lost source or transport event, which results in a dose to a member of the public of greater than 5 Gy (500 rad) (i.e. one with a high probability of early death).**

As a result of the actual release, off-site protective actions are generally unlikely, except for possible local food controls. Other actions can nevertheless be taken as a precaution against further degradation of the plant's status. Plant status is taken into account in the other areas of impact (on-site impact and impact on defence in depth).

*Level 3. Very small release*

Definition: **An external release of radioactivity resulting in a dose (as defined in Section III–1.3) to the critical group of the order of tenths of a millisievert (10mr) or an event, such as a lost source or transport event, which results in a dose to a member of the public leading to acute health effects (such as whole body exposure of the order of 1 Gy (100 rad) and body surface exposure of the order of 10 Gy (1000 rad)).**

Following such an actual release, off-site protection measures are not needed. Such measures can nevertheless be taken as a precaution against further degradation of the plant's status. Plant status is taken into account in the other areas of impact (on-site impact and impact on defence in depth).

Note 1: Radiological equivalence is defined in Section III-1.3.

**III–1.3. Calculation of radiological equivalence and dose**

For levels 5–7, food banning is likely to be implemented and therefore the relative radiological significance of a release to the atmosphere should be assessed by comparing the total committed effective dose from all nuclides resulting from inhala-tion, from the external dose from the passage of the cloud of active material and from the long term external irradiation of deposited activity, i.e. from all pathways except ingestion. Using the assumptions given in Appendix I, the multiplication factor for a range of isotopes has been calculated and is given in Table I. The actual activity released should be multiplied by the factor given and then compared with the values given in the definition of each level.

For levels 3 and 4, there is likely to be little or no food banning, the relative radiological significance is assessed by comparing the committed effective dose for intakes by all routes to the critical group. This should be calculated using the standard national assumptions for dose assessment without taking account of the wind direction at the time of the release or the time of year at which the release occurred. It is not possible to give multiplication factors for levels 3 and 4 as the dose via ingestion will depend on the local agricultural practices.

TABLE I. RADIOLOGICAL EQUIVALENCE FOR OFF-SITE IMPACT *(this applies to levels 5-7 only)*

| Isotope | Multiplication Factor | Isotope | Multiplication Factor | Isotope | Multiplication Factor |
|---|---|---|---|---|---|
| H-3 | 0.02 | Sr-90 | 10 | U-238(M) | 300 |
| I-131 | 1 | Ru-106 | 7 | U-238(F) | 50 |
| Cs-137 | 30 | U-235(S) | 800 | U-Natural | 800 |
| Cs-134 | 20 | U-235(M) | 300 | Pu-239(Class Y) | 10,000 |
| Te-132 | 0.3 | U-235(F) | 100 | Am-241 | 9000 |
| Mn-54 | 4 | U-238(S) | 700 | Noble gasses | Negligible |
| Co-60 | 50 | | | | |
| Note: Lung absorption types: S-slow; M-medium; F-fast. If unsure, use the most conservative value. | | | | | |

Liquid discharges resulting in critical group doses significantly higher than that appropriate for level 4 would need to be rated at level 5 or above but again, the assessment of radiological equivalence would be site specific and therefore detailed guidance cannot be provided here.

## III–2. ON-SITE IMPACT

### III–2.1. General description

The rating of events under on-site impact takes account of the actual impact within the site of the nuclear installation, regardless of the possible off-site releases and defence in depth implications. It considers the extent of major radiological damage, for example core damage, the spread of radioactive products within the site but outside their as-designed containments and the levels of doses to workers.

Events resulting in radiological damage are rated at levels 4 and 5, events resulting in contamination are rated at levels 2 and 3 and events resulting in high doses to workers are rated at levels 2–4. The significance of contamination is measured either by the quantity spread or the resultant dose rate. These criteria relate to dose rates in an operating area but do not require that a worker was actually present. They should not be confused with the criteria for doses to workers which relate to doses actually received.

It is accepted that the exact nature of damage to plant may not be known for some time following an accident with on-site consequences of this nature. However, it should be possible to estimate in broad terms the likelihood of major or minor damage and to decide whether to rate an event provisionally at level 4 or 5 on the scale. It is possible that subsequent re-evaluation of the state of the plant would necessitate re-rating of the event.

Below level 2, on-site impact is considered as insignificant for the purpose of rating an event on the scale; it is only the impact on defence in depth which has to be considered at these lower levels.

### III–2.2. Definition of levels

*Level 5. Severe damage to the reactor core or radiological barriers*

Definition: **More than a few per cent of the fuel in a power reactor is molten or more than a few per cent of the core inventory has been released from the fuel assemblies. Incidents at other installations involving a major release of radioactivity on the site (comparable with the release from a core melt) with a serious off-site radiological safety threat.**

Examples of non-reactor accidents would be a major criticality accident, or a major fire or explosion releasing large quantities of activity within the installation.

Recommended Change:

1. More than 20 per cent of the fuel gap in a power reactor has been released into the reactor coolant and subsequently into the containment from the fuel assemblies. Incidents at other installations involving a major release of radioactivity on the site (comparable with a major release from the fuel clad gap) with a serious off-site radiological safety threat.

Change Justification:

A major release of radioactivity requiring offsite protective actions is not possible unless the containment barrier fails subsequent to a major failure of fuel cladding allowing radioactive material to be released from the core into the reactor coolant. 20 per cent fuel gap release is a value which indicates severe fuel damage. Regardless of whether containment is challenged, this amount of activity in containment, if released, could have such severe consequences that it is prudent to treat this as a potential loss of containment. NUREG-1228, "Source Estimations During Incident Response to Severe Nuclear Power Plant Accidents," indicates that such conditions do not exist when the amount of clad damage is less than 20%. This definition is consistent with the Emergency Action Level (EAL) classification methodology of NEI 99-01, Revision 4, for a General Emergency. Short-term, the evaluation of whether the activity release is a result of damaged clad due to fuel melting is irrelevant and would require either non-ALARA sampling/analysis and/or possible visual fuel inspection to determine.

*Level 4. Significant damage to the reactor core or radiological barriers or fatal exposure of a worker*

Definition: **Any fuel melting has occurred or more than about 0.1% of the core inventory of a power reactor has been released from the fuel assemblies.**

**Events at non-reactor installations involving the release of a few thousand terabecquerels of activity from their primary containment 2 which cannot be returned to a satisfactory storage area.**

**External irradiation of one or more workers, which results in a dose greater than 5 Gy (i.e. one with a high probability of early death).**

Recommended Change:

2. More than a few per cent of the fuel gap (reactor coolant activity >300 μc/cc DEI) in a power reactor has been released into the reactor coolant and subsequently into the containment from the fuel assemblies. Events at non-reactor

installations involving the release of a few thousand terabecquerels (8.1e4 Ci) of activity from their primary containment which cannot be returned to a satisfactory storage area.

Change Justification:

A release of radioactivity requiring on-site protective actions from core damage is not possible unless the containment barrier fails subsequent to a partial failure of fuel cladding allowing radioactive material to be released from the core into the reactor coolant. 5 per cent fuel gap release (reactor coolant activity >300 μc/cc DEI) is a concentration indicative of fuel damage several times larger than the maximum fuel leakage (including iodine spiking) allowed within technical specifications and is therefore indicative of significant fuel damage. This definition is consistent with the Emergency Action Level (EAL) classification methodology of NEI 99-01, Revision 4, for a Site Area Emergency. Escalation to level 5 would occur should activity levels rise to a 20% value. Short-term, the evaluation of whether the activity release is a result of damaged clad due to fuel melting is irrelevant and would require either non-ALARA sampling/analysis and/or possible visual fuel inspection to determine.

*Level 3. Severe spread of contamination and/or overexposure of a worker resulting in acute health effects*

Definition: **Events resulting in a dose rate or a contamination level which did or easily could have resulted in one or more workers receiving a dose leading to acute health effects (such as whole body exposures of the order of 1 Gy (100 rad) and body surface exposures of the order of 10 Gy (1000 rad)).3**
**Events resulting in the release of a few thousand terabecquerels of activity into a secondary containment (see footnote 2) where the mate-rial can be returned to a satisfactory storage area.**

Recommended Change:

3. More than a few per cent of the fuel gap (reactor coolant activity >300 μc/cc DEI) in a power reactor has been released into the reactor coolant from the fuel assemblies. Events resulting in a release of a few thousand terabecquerels (8.1e4 Ci) of activity into a secondary containment where the material can be returned to a satisfactory storage area.

Change Justification:

A release of radioactivity requiring on-site protective actions from core damage is not possible unless a partial failure of fuel cladding allows radioactive material to be released from the core into the reactor coolant. 5 per cent fuel gap release (reactor coolant activity >300 μc/cc DEI) is a concentration indicative of fuel damage several times larger than the maximum fuel leakage (including iodine spiking) allowed within technical specifications and is therefore indicative of fuel damage. With the fuel activity contained within the reactor coolant system, contamination spread may be controlled and activity levels may be reduced through installed isolation and cleanup systems. This definition is consistent with the Emergency Action Level (EAL) classification methodology of NEI 99-01, Revision 4, for an Alert Emergency. Escalation to level 4 would occur should significant reactor coolant leakage into containment subsequently occur.

*Level 2. Major spread of contamination and/or overexposure of workers*

Definition: **Events resulting in a dose to one or more workers exceeding a statutory annual dose limit for radiation workers.**
**Events resulting in the sum of gamma plus neutron dose rates of greater than 50 mSv per hour (5000 mr per hour) in a plant operating area (dose rate measured 1 m from the source).**
**Events leading to the presence of significant quantities of radioactivity in the installation, in areas not expected by design (see the definitions at the end of Part IV) and which require corrective action. In this**
**context 'significant quantity' should be interpreted as:**

(a) Contamination by liquids involving a total activity radiologically equivalent to a few hundred gigabecquerels of $^{106}$Ru.

(b) A spillage of solid radioactive material of radiological significance equivalent to the order of a few hundred gigabecquerels of $^{106}$Ru, providing the surface and airborne contamination levels exceed ten times those permitted for operating areas (see the definitions at the end of Part IV).

(c) A release of airborne radioactive material, contained within a building and involving quantities of radiological significance equivalent to the order of a few tens of gigabecquerels of $^{131}$I.

### III–2.3. Calculation of radiological equivalence

The assumptions to be used in calculating radiological equivalence for on-site impact are given in Appendix I. On the basis of these assumptions, the multiplying factor for a range of isotopes has been calculated and is given in Table II. The actual activity released should be multiplied by the factor given and then compared with the values given in the definition of each level for either $^{131}$I or $^{106}$Ru.

TABLE II. RADIOLOGICAL EQUIVALENCE FOR ON-SITE IMPACT

| Isotope | Multiplication Factor for I-131 equivalence | Multiplication Factor for Ru-106 equivalence |
|---|---|---|
| H-3 | 0.002 | 0.0006 |
| I-131 | 1 | 0.3 |
| Cs-137 | 0.6 | 0.2 |
| Cs-134 | 0.9 | 0.3 |
| Te-132 | 0.3 | 0.1 |
| Mn-54 | 0.1 | 0.03 |
| Co-60 | 1.5 | 0.5 |
| Sr-90 | 7 | 2 |
| Ru-106 | 3 | 1 |
| U-235(S) | 600 | 700 |
| U-235(M) | 200 | 200 |
| U-235(F) | 50 | 20 |
| U-238(S) | 500 | 30 |
| U-238(M) | 100 | 170 |
| U-238(F) | 50 | 20 |
| U-Natural | 600 | 200 |
| Pu-239(Class Y) | 9000 | 3000 |
| Am-241 | 2000 | 700 |
| Noble gasses | Negligible (effectively 0) | Negligible (effectively 0) |
| Note: Lung absorption types: S-slow; M-medium; F-fast. If unsure, use the most conservative value. | | |

# Part IV

# IMPACT ON DEFENCE IN DEPTH

This part of the manual is divided into three main sections. The first gives the background to what is meant by defence in depth. This will probably be familiar to most readers. The second section gives the general principles that are to be used to rate events under defence in depth. As they need to cover a wide range of types of installations and events, they are general in nature. In order to ensure that they are applied in a consistent manner, Section 3 gives more detailed guidance. The guidance is further expanded in Part V, which gives specific guidance for certain types of events and provides a number of worked examples.

## IV–1. BACKGROUND

The avoidance of radiological accidents and incidents, and hence the safety of a nuclear installation, is based on good design and operation. A defence in depth approach is generally applied to both of these aspects and allowance is made for the possibility of equipment failure, human error and the occurrence of unplanned developments.
The definition of defence in depth by the International Nuclear Safety Advisory Group is as follows:

> "To compensate for potential human and mechanical failures, a defence in depth concept is implemented, centred on several levels of protection including successive barriers preventing the release of radioactive material to the environment. The concept includes protection of the barriers by averting damage to the plant and to the barriers themselves. It includes further measures to protect the public and the environment from harm in case these barriers are not fully effective."[4]

Similar defence in depth provisions are provided at all nuclear installations and for the transport of radioactive material. They cover protection of the public and the workforce, and include the means to prevent the transfer of material into poorly shielded locations as well as to prevent radioactive release. Defence in depth is, therefore, a combination of conservative design, quality assurance, surveillance activities, mitigative measures and a general safety culture that strengthens each of the successive layers.

Safe operation is maintained by the three basic safety functions:
- (a) Controlling the reactivity or the process conditions,
- (b) Cooling the radioactive material,
- (c) Confining the radioactive material.

Each of the safety functions is assured by good design, well controlled operation and a range of systems and administrative controls. Within the safety justification for the plant, operational systems may be distinguished from safety provisions; if operational systems fail, then additional safety provisions will operate so as to maintain the safety function. Safety provisions can either be procedures, administrative controls or passive or active systems, which are usually provided in a redundant way, with their availability controlled by operational limits and conditions (OL&C).

The frequency of challenge of the safety provisions is minimized by good design, operation, maintenance, surveillance, etc. For example, the frequency of failures of the primary circuit of a reactor is minimized by design margins, quality control, operational constraints, surveillance, and so on. Similarly, the frequency of reactor transients is minimized by operational procedures, control systems, etc. Normal operational and control systems contribute to minimizing the frequency of challenges to safety provisions.

In some situations it is not possible to reduce significantly the frequency of the challenge of safety provisions, for example attempted entry into cells potentially containing sources. In these cases the safety functions are assured solely by safety provisions of appropriate integrity.

## IV–2.GENERAL PRINCIPLES FOR THE RATING OF EVENTS

This guidance is for application to a wide range of nuclear installations and the radioactive inventory and time-scales of events at such installations will vary widely. These are important factors to be taken into account in rating events and it is inevitable that the guidance here is general and that judgement must be applied. More specific guidance is given in the later sections.

Although three levels for impact on defence in depth are available above level 0, for some installations the maximum possible on-site or off-site consequences are limited by the radioactive inventory and the release mechanism. Clearly the maximum possible level with respect to impact on the defence in depth, where an accident has been prevented, should be lower than the maximum possible level with respect to on-site or off-site impact. If the maximum possible on-site or off-site level for a particular activity cannot be greater than level 4 on the scale because of the limited potential consequences, a maximum rating of level 2 is appropriate under defence in depth. Similarly, if the maximum potential level cannot exceed level 2, then the maximum under defence in depth is level 1.

One facility can, of course, cover a number of activities and each activity must be considered separately in this context. For example, waste storage and reactor operations should be considered as separate activities, even though they can both occur at one facility.

Having identified the upper limit to the rating under defence in depth, the approach to rating is based on assessing the likelihood that the event could have led to an accident, not by using probabilistic techniques directly but by considering whether safety provisions were challenged and what additional failures of safety provisions would be required to result in an accident. Consideration is also given as to whether any underlying cultural issues are evident in the event that might have increased the likelihood of the event leading to an accident.

The following steps should therefore be followed to rate an event:

(1) The upper limit to the rating under defence in depth should be established by taking account of the maximum potential radiological consequences (i.e. the maximum potential rating for the relevant activities at that facility under off-site and on-site impact). Further guidance on establishing the maximum potential consequences is given in Section IV–3.1.

(2) The basic rating should then be determined by taking account of the number and effectiveness of the safety provisions available (hardware and administrative) for prevention, surveillance and mitigation, including passive and active barriers. In identifying the number and effectiveness of such provisions it is important to take account of the time available and the time required for identifying and implementing appropriate corrective action. Further guidance on the assessment of safety provisions is provided in Section IV–3.2.

(3) In addition to the above considerations, increasing the basic rating should be considered, as explained in Section IV–3.3, within the upper limit of the defence in depth rating established in item (1) above. Uprating allows for those aspects of the event that may indicate a deeper degradation of the plant or the organizational arrangements of the facility. Factors considered are common cause failures, procedural inadequacies and safety culture deficiencies. Such factors are not included in the basic rating and may indicate that the significance of the event with respect to defence in depth is higher than the one considered in the basic rating process. Accordingly, in order to communicate the true significance of the event to the public, uprating by one level is considered. Clearly, as well as considering the event under defence in depth, each event must also be considered against off-site and on-site impact.

## IV–3. DETAILED GUIDANCE FOR RATING EVENTS

### IV–3.1. Identification of maximum potential consequences

For the assessment of events affecting the majority of the reactor core or the fuel in the spent fuel pool of power reactors, it is generally not necessary to specifically consider the maximum potential consequences. The theoretical possibility of a large release is recognized and therefore the upper limit to the rating under defence in depth is level 3.

For other facilities, or for activities involving only a small fraction of the core inventory (e.g. fuel handling), it is necessary to consider the maximum potential consequences (i.e. the maximum potential rating under off-site and on-site impact) should all the safety provisions fail. For some facilities it may not be physically possible to reach the upper levels of INES even from extremely unlikely accidents. The maximum potential consequences are not specific to the type of event but apply to a set of operations at a facility.

In assessing the maximum potential rating under off-site and on-site impact, the following general principles should be taken into account:

(a) Any one site may contain a number of facilities with a range of tasks carried out at each facility. Thus the maximum potential rating should be specific to the type of facility at which the event occurred and the type of operations being undertaken at the time of the event.

(b) It is necessary to consider both the radioactive inventory that could potentially have been involved in the event, the physical and chemical properties of the material involved, and the mechanisms by which that activity could have been dispersed.
(c) The consideration should not focus on the scenarios considered in the safety justification of the plant but should consider physically possible accidents had all the plant safety provisions threatened by the event been deficient.

These principles can be illustrated by the following examples:

(1) For events associated with maintenance cell entry interlocks, the maximum potential consequences are likely to be related to worker exposure. If the radiation levels are sufficiently high to cause worker death if the cell is entered and no mitigative actions are taken, then the maximum potential rating is at level 4 under on-site impact.
(2) For events involving small research reactors (i.e. with power less than 1 MW), although the physical mechanisms exist for the dispersal of a significant fraction of the inventory (either through criticality accidents or loss of fuel cooling), the total inventory is such that the maximum potential rating could not be higher than level 4, either on-site or off-site, even if all the safety provisions fail.
(3) For reprocessing facilities and other facilities processing plutonium com-pounds, the inventory and physical mechanisms which exist for the dispersal of a significant fraction of that inventory (either through criticality accidents, chemical explosions or fires), are such that the maximum potential rating could exceed level 4, either under off-site or on-site impact, if all the safety provisions fail.
(4) For uranium fuel fabrication and enrichment plants, releases have chemical and radiological safety aspects. It has to be emphasized that the chemical risk posed by the toxicity of fluorine and uranium predominates over the radiological risk. INES, however, is only related to the assessment of the radiological hazard. From a radiological standpoint, no severe off-site or on-site consequences exceeding a rating of level 4 are conceivable from a release of uranium or its compounds.

## IV–3.2. Identification of basic rating taking account of the effectiveness of safety provisions

Because the safety analysis for reactor installations during power operation follows a common international practice, it is possible to give more specific guidance about how to assess the safety provisions for events involving reactors at power. In addition, as noted at the start of Section IV–3.1, the rating does not need to explicitly consider the maximum potential consequences. The approach is based on consideration of **initiators, safety functions** and **safety systems.** These terms will be familiar to those involved in safety analysis but further explanation of the terms is provided below. Other events at reactor sites, e.g. those associated with a shutdown reactor or with other facilities on the site, should be rated using the safety layers approach described in Section IV–3.2.2. Similarly, events involving research reactors should use the safety layers approach to take proper account of maximum potential consequences and design philosophy. An overview of the approach to help those new to the scale is given in Appendix II.

### *IV–3.2.1. Events occurring on reactors at power (initiator approach)*

An initiator or initiating event is an identified event that leads to a deviation from the normal operating state and challenges one or more safety functions. Initiators are used in safety analysis to evaluate the adequacy of installed safety systems: the initiator is an occurrence that challenges the safety systems and requires them to function.

Events involving an impact on the plant defence in depth will generally be of two possible forms:

—Either an initiator (initiating event) which requires the operation of some particular safety systems designed to cope with the consequences of this initiator;
—Or degraded operability of a safety function owing to the operability of one or more safety systems being degraded without the occurrence of the initiator for which the safety systems had been provided.

In the first case, the event rating depends mainly on the extent to which the operability of the safety function is degraded. However, the severity also depends on the anticipated frequency of the particular initiator.

In the second case, no deviation from normal operation of the plant actually occurs, but the observed degradation of the operability of the safety function could have led to significant consequences if one of the initiators for which the degraded safety systems are provided had actually occurred. In such a case, the event rating again depends on:
—The anticipated frequency of the potential initiator,

—The operability of the associated safety function assured by the operability of particular safety systems.

It has to be pointed out that one particular event could be categorized under both cases.

The basic approach to rating such events is therefore to identify the frequency of the relevant initiators and the operability of the affected safety functions. Two tables are then used to identify the appropriate basic rating. Further information on the derivation of the tables is given in Appendix III. Detailed guidance on rating is given below.

IV–3.2.1.1. Identification of initiator frequency

Four different frequency categories have been selected:

(1) *Expected.* This covers initiators expected to occur once or several times during the operating life of the plant.
(2) *Possible.* Initiators which are not 'expected', but have an anticipated frequency during the plant lifetime of greater than about 1% (i.e. about $3 \times 10^{-4}$ per year).
(3) *Unlikely.* Initiators considered in the design of the plant which are less likely than the above.
(4) *Beyond design.* Initiators of very low frequency, not normally included in the conventional safety analysis of the plant. When protection systems are introduced against these initiators, they do not necessarily include the same level of redundancy or diversity as measures against design basis accidents.

Each plant has its own list and classification of initiators. Typical examples of design basis initiators categorized into the previous classes are given in Appendix IV. Small plant perturbations that are corrected by control (as opposed to safety) systems are not included in the initiators. The initiator may be different from the occurrence which starts the event; on the other hand a number of different event sequences can often be grouped under a single initiator.

For many events, it will be necessary to consider more than one initiator, each of which will lead to a rating. The event level will be the highest of the levels associated with each initiator. For example, a power excursion in a reactor could be an initiator challenging the protection function. Successful operation of the protection system would then lead to a shutdown. It would then be necessary to consider the reactor trip as an initiator challenging the fuel cooling function.

IV–3.2.1.2. Safety function operability

The three basic safety functions are:

(a) Controlling the reactivity or the process conditions,
(b) Cooling the radioactive material,
(c) Confining the radioactive material.

These functions are provided by passive systems (such as physical barriers) and active systems (such as the reactor protection system). Several safety systems may con-tribute to a particular safety function, and the function may still be achieved even with one system unavailable. Equally, support systems such as electrical supplies, cooling and instrument supplies will be required to ensure that a safety function is achieved. It is important that it is the operability of the safety function that is considered when rating events, not the operability of an individual system. A system or component shall be considered operable when it is capable of performing its required function in the required manner.

Operational limits and conditions govern the operability of each safety system. In most countries they are included within the Technical Specifications.

The operability of a safety function for a particular initiator can range from a state where all the components of the safety systems provided to fulfil that function are fully operable to a state where the operability is insufficient for the safety function to be achieved. To provide a framework for rating events, four categories of operability are considered.

**A. Full**

All safety systems and components which are provided by the design to cope with the particular initiator in order to limit its consequences are fully operable (i.e. redundancy/diversity is available).

**B. Minimum required by OL&C**

The minimum operability of safety systems providing the required safety function specified in OL&C for which continued operation at power is permitted, even for a limited time. This level of operability will generally correspond to the minimum operability of the different safety systems for which the safety function can be achieved for all the initiators considered in the design of the plant. However, for certain particular initiators redundancy and diversity may still exist.

**C. Adequate**

A level of operability of safety systems sufficient to achieve the particular safety function for the initiator being considered. For some safety systems, this will correspond to a level of operability lower than that required by OL&C. An example would be where diverse safety systems are each required to be operable by OL&C, but only one is operable, or where all safety systems which are designed to assure a safety function are inoperable for such a short time that the safety function, although outside OL&C, is still assured by other means (for example, the safety function 'cooling of the fuel' may be assured if a total station blackout occurs for only a short time). In other cases, categories B and C may be the same.

**D. Inadequate**

The degraded operability of the safety systems is such that the safety function cannot be fulfilled for the initiator being considered.

It should be noted that although C and D represent a range of plant states, A and B represent specific operabilities. Thus the actual operability may be between that defined by A and B, i.e. the operability may be less than full but more than the minimum allowed for continued operation at power. This is considered in Section IV–3.2.1.3(a).

### IV–3.2.1.3. Assessment of the basic rating

In order to obtain a basic categorization, first decide whether there was an actual challenge to the safety systems (a real initiator). If so, then Section IV–3.2.1.3(a) is appropriate, otherwise Section IV–3.2.1.3(b) is appropriate. It may be necessary to consider an event using both sections if an initiator occurs and reveals a reduced operability in a function not challenged by the real initiator, e.g. if a reactor trip without loss of off-site power reveals a reduced operability of diesels. For events involving potential failures, e.g. discovery of structural defects, a similar approach is used as described in Section IV–3.2.3.

**(a) Events with a real initiator**

The first step is to decide the frequency with which that type of initiator was expected by design. In deciding the appropriate category, it is the frequency that was assumed in the safety case (the justification of the safety of the plant and its operating envelope) for the plant that is relevant. Appendix IV provides some examples.

The second step is to determine the operability of the safety functions challenged by the initiator. It is important that only those safety functions challenged are considered. If the degradation of other safety systems is discovered, it should be assessed using Section IV–3.2.1.3(b) against the initiator that would have challenged that safety function. It is also important to note that in deciding whether the operability is within OL&C, it is the operability requirements prior to the event that must be considered, not those that apply during the event. If the operability is within OL&C but also just adequate, category C should be used.

The event rating should then be determined from Table III. Where a choice of rating is given, the choice should be based on the extent of redundancy and diversity available for the initiator being considered. If the safety function operability is just adequate (i.e. one further failure would have lead to an accident), level 3 is appropriate.
In cell B1 of Table III, the lower value would be appropriate if there is still considerable redundancy and/or diversity available.

Where the safety function operability is greater than the minimum required by OL&C, but less than 'Full', there may be considerable redundancy and diversity available for expected initiators. In such cases, level 0 would be more appropriate.

Beyond design initiators are not included specifically in Table III. If such an initiator occurs, then levels 2 or 3 are appropriate under defence in depth depending on the redundancy of the systems providing protection. However, it is possible that beyond design initiators will lead to an accident requiring classification under off-site or on-site impacts.

The occurrence of internal and external hazards such as fires, external explosions or tornadoes may be rated using the table. The hazard itself should not be considered as the initiator, but the safety systems that remain operable should be assessed against an initiator that occurred and/or against potential initiators.

TABLE III. EVENTS WITH A REAL INITIATOR

| Initiator Frequency | | Expected | Possible | Unlikely |
|---|---|---|---|---|
| Safety function operability | | | | |
| A | Full | 0 | 1 | 2 |
| B | Within OL&C | 1/2 | 2/3 | 2/3 |
| C | Adequate | 2/3 | 2/3 | 2/3 |
| D | Inadequate | 3+ | 3+ | 3+ |

**(b) Events without a real initiator**

The first step is to determine the safety function operability. In practice, safety systems or components may be in a state not fully described by any of the four categories. The operability may be less than full but more than the minimum required by OL&C, or the whole system may be available but degraded by loss of indications. In such cases the relevant categories should be used to give the possible range of the rating, and judgement used to determine the appropriate rating. If the operability is just adequate but still within OL&C, category B should be used.

The second step is to determine the frequency of the initiator for which the safety function is required. If there is more than one relevant initiator, then each must be considered. The one giving the highest rating should be used. If the frequency lies on the boundary between two categories some judgement will need to be applied. For systems specifically provided for protection against hazards, the hazard should be considered as the initiator.

The event rating should then be determined from Table IV. Where a choice of rating is given, the choice should be based on whether the operability is just adequate or whether redundancy and/or diversity still exists for the initiator being considered. If the period of inoperability was very short compared with the interval between tests of the components of the safety system, consideration should be given to reducing the basic rating of the event.

Beyond design initiators are not included specifically in Table IV. Where the operability of the affected safety function is less than the minimum required by OL&C, level 1 is appropriate. If the operability is greater than the minimum required by OL&C, or OL&C do not provide any limitations on the system operability, level 0 is appropriate.

TABLE IV. EVENTS WITHOUT A REAL INITIATOR

| Initiator Frequency | | Expected | Possible | Unlikely |
|---|---|---|---|---|
| Safety function operability | | | | |
| A | Full | 0 | 0 | 0 |
| B | Within OL&C | 0 | 0 | 0 |
| C | Adequate | 1/2 | 1 | 1 |
| D | Inadequate | 3 | 2 | 1 |

*IV–3.2.2. All other events, i.e. any event not associated with reactors at power (the layers approach)*

*IV–3.2.3. Potential events (including structural defects)*

Some events do not of themselves challenge the safety provisions but do correspond to an increased likelihood of a challenge. Examples are the discovery of structural defects, a leak terminated by operator action or faults discovered in process control systems. The approach to rating such events is described below.

The surveillance programme is intended to identify structural defects before their size becomes unacceptable. If the defect is within this size, then level 0 would be appropriate. If the defect is larger than expected under the surveillance programme, categorization of the defects needs to take account of two factors.

First, the safety significance of the defective component should be deter-mined by assuming that the defect had led to failure of the component and applying the appropriate part of Section IV–3. If using Section IV–3.2.1 (reactors at power), then if the defect is in a safety system, applying Section IV–3.2.1.3(b) will give the upper limit of the basic rating. The possibility of common mode failure may need to be considered. If the defect was in a component whose failure could have led to an initiator, then applying Section IV–3.2.1.3(a) will give the upper value of the basic rating.

The potential rating derived in this way should then be adjusted depending on the likelihood that the defect would have led to component failure, and by consideration of the additional factors discussed in Section IV–3.3.

Other potential events can be assessed in a similar way to that described above. First, the significance of the potential challenge should be evaluated by assuming that it had actually occurred and applying the appropriate part of Section IV–3, based on the operability of safety provisions that existed at the time. Secondly, the rating should be reduced, depending on the likelihood that the potential challenge could have developed from the event that actually occurred. The level to which the rating should be reduced must be based on judgement.

*IV–3.2.4. Events rated below scale at level 0*

In general, events should be classified below scale at level 0 only if application of the procedures described above does not lead to a higher rating. However, provided none of the additional factors discussed in Section IV–3.3 are applicable, the following types of event are typical of those that will be categorized as below scale at level 0:

—Reactor trip proceeding normally;
—Spurious operation of the safety systems [6] followed by normal return to operation without affecting the safety of the installation;
—No significant degradation of the barriers (leak rate less than OL&C);
—Single failures or component inoperability in a redundant system discovered during scheduled periodic inspection or test.

## IV–3.3. Consideration of additional factors

Particular aspects may challenge simultaneously different layers of the defence in depth and are consequently to be considered as additional factors which may justify an event having to be classified one level above the one resulting from the previous guidance.

The main additional factors which act in such a way are:

—Common cause failures,
—Procedural inadequacies,
—Safety culture deficiencies.

Because of such factors, it may happen that an event could be rated at level 1, although of no safety significance on its own, without taking those additional factors into account.

[6] Spurious operation in this respect would include operation of a safety system as a result of a control system malfunction, instrument drift or individual human error. However, the actuation of the safety system initiated by variations in physical parameters which have been caused by unintended actions elsewhere in the plant would not be considered as spurious initiation of the safety system.

When considering the upgrading of the basic level based on the above factors, the following aspects require consideration:

(1) Some of the above factors may have already been included in the basic rating, e.g. common mode failure. It is therefore important to take care that such failures are not double counted. Allowing for all additional factors, the level of an event can only be upgraded by one level.
(2) The event should not be uprated beyond the maximum level derived in accordance with Section IV–2, and this maximum level should only be applied if, had one other event taken place (either an expected initiator or a further component failure), an accident would have occurred.

*IV–3.3.1. Common cause failures*

A common cause failure is the failure of a number of devices or components to perform their functions as a result of a single specific event or cause. In particular, it can cause the failure of redundant components or devices intended to perform the same safety function. This may imply that the reliability of the whole safety function could be much lower than expected. The severity of an event which implies a common cause failure affecting one or several components is therefore higher than a random failure affecting the same components.

Events where there is a difficulty in operating systems caused by missing or misleading information can also be considered for uprating on the basis of a common cause failure.

*IV–3.3.2. Procedural inadequacies*

The simultaneous challenge of several layers of defence in depth may arise because of inadequate procedures. Such inadequacies are therefore also a possible reason for uprating the level on the scale. Examples include: incorrect or inadequate instructions given to operators for coping with an event (during the Three Mile Island accident in 1979, the procedures to be used by operators in the case of safety injection actuation were not adapted for the particular situation of a loss of coolant in the steam phase of the pres-surizer); or deficiencies in the surveillance programme highlighted by anomalies not dis-covered by normal procedures or plant unavailabilities well in excess of the test interval.

*IV–3.3.3. Events with implications for safety culture*

Safety culture has been defined as "that assembly of characteristics and attitudes in organizations and individuals which establishes that, as an overriding priority, nuclear plant safety issues receive the attention warranted by their significance". A good safety culture helps to prevent incidents but, on the other hand, a lack of safety culture could result in operators performing in ways not in accordance with the assumptions of the design. Safety culture has therefore to be considered as part of the defence in depth and consequently, a deficiency in safety culture could justify upgrading the rating of an event by one level.

To merit upgrading due to a deficiency in the safety culture, the event has to be considered as a real indicator of a deficiency in the overall safety culture.

Examples of such indicators could be:

—A violation of operational limits and conditions or a violation of a procedure without justification (see Appendix V for additional information on OL&C and Technical Specifications);
—A deficiency in the quality assurance process;
—An accumulation of human errors;
—A failure to maintain proper control over radioactive materials, including releases into the environment or a failure in the systems of dose control;
—The repetition of an event, indicating that either the possible lessons have not been learnt or the corrective actions have not been taken after the first event.

**It is important to note that the intention of this guidance is not to initiate a long and detailed assessment but to consider if there is an immediate judgement that can be made by those rating the event.**

IV–4. DEFINITIONS

This section provides definitions for words not defined in other IAEA publications. In many cases a more detailed explanation is provided in this manual.

**areas not expected by design.** Areas whose design basis, for either permanent or temporary structures, does not assume that following an incident the area could receive and retain the level of contamination that has occurred and prevent the spread of contamination beyond the area. Examples of events involving contamination of areas not expected by design, are:

—Contamination by radionuclides outside controlled or supervised areas, where normally no activity is present like floors, staircases, auxiliary buildings, storage areas, etc.
—Contamination by plutonium or highly radioactive fission products of an area designed and equipped only for the handling of uranium.

**authorized operating regime.** See operating limits and conditions.
**defence in depth.** As defined in 'Basic Safety Principles for Nuclear Power Plants' (Safety Series No. 75-INSAG-3 Rev. 1) (see footnote 4):

"To compensate for potential human and mechanical failures, a defence in depth concept is implemented, centred on several levels of protection including successive barriers preventing the release of radioactive material to the environment. The concept includes protection of the barriers by averting damage to the plant and to the barriers themselves. It includes further measures to protect the public and the environment from harm in case these barriers are not fully effective."

**high integrity safety layer.** Should have all of the following characteristics:
(a) The safety layer is designed to cope with all relevant design basis faults and is explicitly or implicitly recognized in the plant safety justification as requiring particularly high reliability or integrity.
(b) The integrity of the safety layer is assured through appropriate monitoring or inspection such that any degradation of integrity is identified.
(c) If any degradation of the layer is detected, there are clear means of coping with the event and of implementing corrective actions, either through pre-deter-mined procedures or through long times being available to repair or mitigate the fault.

**initiator (initiating event).** An identified event that leads to a deviation from thenormal operating state and challenges one or more safety functions.

**operability of a safety function.** The operability of a safety function can be 'Full', 'Within OL&C', 'Adequate' or 'Inadequate', depending upon the operability of the individual redundant and diverse safety systems and components.

**operability of equipment.** A component shall be considered operable when it is capable of performing its required function in the required manner.

**operating area.** Areas where worker access is permitted. It excludes areas where specific controls are required owing to the level of contamination or radiation.

**operational limits and conditions (OL&C).** A set of rules which set forth parameter limits, the functional capability and the performance levels of equipment and personnel approved by the regulatory body for safe operation of the nuclear power plant (in most countries, these are included within 'Technical Specifications').

**radiological barrier.** A barrier designed to prevent dispersion of radioactive material beyond its intended containment.

**radiological equivalence.** The quantity of a radionuclide which must be released to give the same committed effective dose as the reference quantities of $^{131}$I or $^{106}$Ru under on-site and off-site impact, calculated using the model detailed in Appendix I.

**safety functions.** The three basic safety functions are: (a) controlling the reactivity or the process conditions; (b) cooling the radioactive material; and (c) confining the radioactive material.

**safety layers.** A safety provision that cannot be broken down into redundant parts.

**safety provisions.** Procedures, administrative controls, or passive or active systems which are usually provided in a redundant way, with their availability controlled by OL&C.

**safety relevance.** Concerns nuclear or radiological safety.

**safety systems.** Systems important to safety, provided to ensure the safety functions.

## Part V

## EXAMPLES TO ILLUSTRATE THE DEFENCE IN DEPTH RATING GUIDANCE

# V–1. GUIDANCE ON THE USE OF THE LAYERS APPROACH FOR SPECIFIC TYPES OF EVENTS

# Part VI

# APPENDICES

## Appendix I

## CALCULATION OF RADIOLOGICAL EQUIVALENCE

## Appendix II

## OVERVIEW OF THE PROCEDURE FOR RATING EVENTS FOR REACTORS AT POWER UNDER DEFENCE IN DEPTH

### II.1. BACKGROUND

Defence in depth can be considered in a number of different ways. For example, one can consider the number of barriers provided to prevent a release (e.g. fuel, clad, pressure vessel, containment). Equally one can consider the number of systems that would have to fail before an accident could occur (e.g. loss of off-site power plus failure of all essential diesels). It is the latter approach that is adopted within the INES rating procedure.

The basic rating procedure concentrates on the extent of safety system failures, and whether they have been challenged. However, it is recognized that the consequences of all the systems failing can vary considerably. Potential consequences are treated within INES in a relatively simple manner. For events where the maximum potential consequences could be level 5 or higher, level 3 is the maximum appropriate under defence in depth. If the maximum potential consequences of the event cannot be greater than level 4, then the maximum under defence in depth is level 2. Similarly, if the maximum potential consequences cannot exceed level 2, then the maximum under defence in depth is level 1.

We will now consider the approach to rating events in more detail. Two separate but similar approaches are described in the manual. The first, which is summarized here, is most obviously appropriate for events associated with reactors at power. The second is more likely to be appropriate for events related to shutdown reactors, chemical plants, fuel route faults, provisions associated with protection to workers, etc. In general, the approach to be used depends upon the manner in which the safety of the plant has been assessed.

### II.2. PROCEDURE FOR EVENTS ASSOCIATED WITH REACTORS AT POWER

Consider a plant where the protection against loss of off-site power is provided by four essential diesels. In order for an accident to occur, the event must challenge plant safety (e.g. LOOP) and the protection must fail (e.g. all diesels fail to start). The initial challenge to plant safety (LOOP in the example) is termed the 'initiator' and the response of the diesels is defined by the 'Operability of the safety function' (post-trip cooling in this example). Thus, for an accident to occur there needs to be an initiator and inadequate operability of safety functions.

Defence in depth measures how near we are to that accident, i.e. whether the initiator has occurred, how likely it was and the operability of the safety functions. If off-site power had been lost but all diesels started as intended, an accident was unlikely (such an event would probably be rated at level 0). Similarly, if one diesel had failed under a test but the others were available and off-site supplies were available, then an accident was unlikely (again such an event would probably be rated at level 0).

However, if it was discovered that all diesels had been unavailable for a month, then even though off-site power had been available and the diesels were not required to operate, an accident was relatively likely as the chance of losing off-site power was relatively high (such an event would probably be rated at level 3 provided there were no other lines of protection).

The rating procedure therefore considers whether the safety functions were required to work (i.e. had an initiator occurred), the assumed likelihood of the initiator and the operability of the relevant safety functions.

# Appendix III

## DERIVATION OF THE TABLES FOR RATING EVENTS FOR REACTORS AT POWER (SECTION IV–3.2.1)

### III.1. INCIDENTS INVOLVING A DEGRADATION OF SAFETY SYSTEMS WITHOUT AN INITIATOR (SECTION IV–3.2.1.3(b))

The categorization of an incident will depend primarily on the extent to which the safety functions are degraded and on the likelihood of the initiator for which they are provided. Strictly speaking, the latter is the likelihood of the initiator occurring during the period of safety function degradation since the period of inoperability will vary from one incident to another. Accordingly, if the period of inoperability is very short, a level lower than that provided in the table may be appropriate.

If the operability of a required safety function is inadequate (no matter if it is just inadequate or very inadequate), then an accident was only prevented because the initiator did not occur. For such an incident, if the safety function is required for expected initiators (i.e. those expected to occur once or more during the life of the plant), level 3 is appropriate. If the inadequate safety function is only required for possible or unlikely initiators, a lower level is clearly appropriate because the likelihood of an accident is much lower. For this reason, the table shows level 2 for possible initiators and level 1 for unlikely initiators.

The level chosen should clearly be less when the safety function is adequate than when it is inadequate. Thus, if the function is required for expected initiators, and the operability is just adequate, level 2 is appropriate. However, in a number of cases the safety function operability may be considerably greater than just adequate, but not within OL&C. This is because the minimum operability required by OL&C will often still incorporate redundancy and/or diversity against some expected initiators. In such situations, level 1 would be more appropriate. Thus, the table shows a choice of level 1 or 2. The appropriate value should be chosen depending on the remaining redundancy and/or diversity.

If the safety function is required for possible or unlikely initiators, then reduction by one from the level derived above for an inadequate system gives level 1 for possible initiators and level 0 for less likely initiators. However, it is not considered appropriate to categorize at level 0 a reduction in safety system operability below that required by the OL&C. One important part of defence in depth, a redundant safety system, has been defeated. Thus, level 1 is shown in the table for both possible and unlikely initiators.

If the safety function operability is within the OL&C the plant has remained within its safe operating envelope and level 0 is appropriate for all frequencies of initiators. This is also shown in the table.

### III.2. INCIDENTS INVOLVING A REAL INITIATOR (SECTION IV–3.2.1.3(a))

Here the categorization will depend primarily on the operability of the safety functions, but for consistency the same table structure as for events without real initiators is used.

Clearly, if the safety function is inadequate, an accident will have occurred and it may be categorized under off-site or on-site impact. However, in terms of defence in depth, level 3 represents the highest category. This total loss of defence in depth is expressed by 3+ in the table.

If the safety function is just adequate, then again level 3 is appropriate, as a further failure would lead to an accident. However, as noted in the previous section, when inoperability is just less than that required by the operational limits and conditions, it may be considerably greater than just adequate, particularly for expected initiators. Therefore, in the table level 2/3 is shown for expected initiators and adequate safety function, the choice depending on the extent to which the operability is greater than just adequate. For unlikely initiators the operability required by the operational limits and conditions is likely to be just adequate and, therefore, in general level 3 would be appropriate for adequate operability. However, there may be particular initiators for which there is redundancy and therefore the table shows level 2/3 for all initiator frequencies.

If there is full safety function operability and an expected initiator occurs, this should clearly be level 0, as shown in the table. However, occurrences of possible or unlikely initiators, even though there may be considerable redundancy in the safety systems, represent a failure of one of the important parts of defence in depth, namely the prevention of initiators. For this reason the table shows level 1 for possible initiators and level 2 for unlikely initiators.

If the operability of safety functions is the minimum required by OL&C, then in some cases, as already noted, for possible and particularly for unlikely initiators, there will be no further redundancy. Therefore, level 2/3 is appropriate, depending on the remaining redundancy. For expected initiators, there will be additional redundancy and therefore a lower categorization is proposed. The table shows level 1/2, where again the value chosen should depend on the additional redundancy within the safety functions. Where the safety function availability is greater than the minimum required by OL&C but less than full, there may be considerable redundancy and diversity available for expected initiators. In such cases, level 0 would be more appropriate.

# Appendix IV

# EXAMPLES OF INITIATORS

## IV.1. PRESSURIZED WATER REACTORS (PWR AND WWER)

### IV.1.1. Expected

—Reactor trip;
—Inadvertent chemical shim dilution;
—Loss of main feedwater flow;
—Reactor coolant system depressurization by inadvertent operation of an active component (e.g. a safety or relief valve);
—Inadvertent reactor coolant system depressurization by normal or auxiliary pressurizer spray cooldown;
—Power conversion system leakage that would not prevent a controlled reactor shutdown and cooldown;
—Steam generator tube leakage in excess of plant Technical Specifications, but less than the equivalent of a full tube rupture;
—Reactor coolant system leakage that would not prevent a controlled reactor shutdown and cooldown;
—Loss of off-site AC power, including consideration of voltage and frequency disturbances;
—Operation with a fuel assembly in any misoriented or misplaced position;
—Inadvertent withdrawal of any single control assembly during refuelling;
—Minor fuel handling incident;
—Complete loss or interruption of forced reactor coolant flow, excluding reactor coolant pump locked rotor

### IV.1.2. Possible

—Small LOCA,
—Full rupture of one steam generator tube,
—Dropping of a spent fuel assembly involving only the dropped assembly,
—Leakage from spent fuel pool in excess of normal make-up capability,
— Blowdown of reactor coolant through multiple safety or relief valves.

### IV.1.3. Unlikely

—Major LOCA, up to and including the largest justified pipe rupture in the reactor coolant pressure boundary;
—Single control rod ejection;
—Major power conversion system pipe rupture, up to and including the largest justified pipe rupture;
—Dropping of a spent fuel assembly onto other spent fuel assemblies.

## IV.2. BOILING WATER REACTORS

### IV.2.1. Expected

—Reactor trip;
—Inadvertent withdrawal of a control rod during reactor operation at power;
—Loss of main feedwater flow;
—Failure of reactor pressure control;
—Leakage from main steam system;
—Reactor coolant system leakage that would not prevent a controlled reactor shutdown and cooldown;
—Loss of off-site power AC, including consideration of voltage and frequency disturbances;
—Operation with a fuel assembly in any misoriented or misplaced position;
—Inadvertent withdrawal of any single control rod assembly during refuelling;
—Minor fuel handling incident;
—Loss of forced reactor coolant flow.

### IV.2.2. Possible

—Small LOCA,

—Rupture of main steam piping,
—Dropping of spent fuel assembly involving only the dropped assembly,
—Leakage from spent fuel pool in excess of normal make-up capability,
—Blowdown of reactor coolant through multiple safety or relief valves.

**IV.2.3. Unlikely**
—Major LOCA, up to and including the largest justified pipe rupture in the reactor coolant pressure boundary;
—Single control rod drop;
—Major rupture of main steam pipe;
—Dropping of a spent fuel assembly onto other spent fuel assemblies.

## Appendix V

## RATING OF EVENTS INVOLVING VIOLATION OF OL&C

The 'operational limits and conditions' describe the minimum operability of safety systems such that operation remains within the safety requirements of the plant. They may also include operation with reduced safety system availability for a limited time. In some countries, 'Technical Specifications' include OL&C and, furthermore, in the event that the OL&C are not met, describe the actions to be taken, including times allowed for recovery and the appropriate fallback state.

If the system availability is within the OL&C but the utility stays more than the allowed time (as defined in the Technical Specification) in that availability state, the event should be rated at level 1 because of deficiencies in safety culture.

If the system availability is discovered to be less than that allowed by the OL&C, even for a limited time, but the operator goes to a safe state in accordance with the Technical Specifications, the event should be rated as described in Section III–3.2, but should not be uprated due to violation of the Technical Specifications. Account should also be taken of the time for which the safety function availability is less than that defined by the OL&C.

In addition to the formal OL&C, some countries introduce into their Technical Specifications further requirements such as limits that relate to the long term safety of components. For events where such limits are exceeded for a short time, level 0 may be more appropriate.

For reactors in the shutdown state, Technical Specifications will again specify minimum availability requirements, but will not generally specify recovery times and fall back states as it is not possible to identify a safer state. The requirement will be to restore the original plant state as soon as possible. In general, plant failures that reduce availability during shutdown should be rated using the safety layers approach and the reduction in plant availability below that required by the Technical Specifications should not be regarded as a violation of OL&C.

This manual was prepared on the basis of experience gained in applying the 1992 edition and the clarification of issues raised. This updating was carried out under the auspices of the INES Advisory Committee, chaired by S. Mortin, Magnox Generation Business Group, British Nuclear Fuels, United Kingdom.

## Appendix VI

## LIST OF PARTICIPATING COUNTRIES AND ORGANIZATIONS

| INES | SAFETY SIGNIFICANCE DESCRIPTION | Current OFF-SITE CRITERIA | Current ON-SITE CRITERIA Core Damage | Proposed ON-SITE CRITERIA Core Damage | Current ON-SITE CRITERIA Radiological Release/Barriers/ Effects | Current DEFENSE IN DEPTH DEGRADATION | Notes |
|---|---|---|---|---|---|---|---|
| 7 | MAJOR ACCIDENT | **Major release widespread health and environmental effects to require implementation of countermeasures.** | | | Release equivalent to > 10,000 terabecquerrels $I_{131}$ (> 2.7e5 curies) (~ 20% total gap) | | |
| 6 | SERIOUS ACCIDENT | **Significant release likely to require full implementation of planned countermeasures** | | | Release equivalent to 1000 to 10,000 terabecquerrels $I_{131}$ (2.7e4 to 2.7e5 curies) | | |
| 5 | ACCIDENT WITH OFF-SITE RISK | **Limited release likely to require partial implementation of planned countermeasures** | **Severe damage to reactor core/radiological barriers** - more than a few % of fuel is molten or more than > a few % of the core inventory has been released from the fuel assemblies (note 1) | More than 20 per cent of the fuel gap in a power reactor has been released into the reactor coolant and subsequently into the containment from the fuel assemblies. | External Release equivalent to 100's to 1,000's of terabecquerrels $I_{131}$ (2.7e3 to 2.7e4 curies) | | Note 1- For non-reactor installations: a major release of radioactivity on the site (comparable with release from a core melt) with serious off-site radiological safety threat |
| 4 | ACCIDENT WITHOUT SIGNIFICANT OFF-SITE RISK | **Minor release with public exposure on the order of prescribed limits** - off-site protective actions generally unlikely except possible local food control | **Significant Damage to reactor core/radiological barriers** - Any fuel melting has occurred or more than about 0.1% of the core inventory has been released from the assemblies (note 2) | More than a few per cent of the fuel gap (reactor coolant activity >300 uc/cc DEI) in a power reactor has been released into the reactor coolant and subsequently into the containment from the fuel assemblies. | Release resulting in dose to the most exposed off-site individual of a few milliseverts (300 mr)<br><br>Event results in worker exposure to 5 Gy (500R) | | Note 2 - For non-reactor installations the release criteria for an INES 4 is a few thousand TBq release from the primary containment (8.1e4 Ci) |

| INES | SAFETY SIGNIFICANCE DESCRIPTION | Current OFF-SITE CRITERIA | Current ON-SITE CRITERIA Core Damage | Proposed ON-SITE CRITERIA Core Damage | Current ON-SITE CRITERIA Radiological Release/Barriers/ Effects | Current DEFENSE IN DEPTH DEGRADATION | Notes |
|---|---|---|---|---|---|---|---|
| 3 | SERIOUS INCIDENT | **Very small release with public exposure at a fraction of prescribed limits -** off-site protective measures may not be needed | **Severe spread of contamination -** Damage resulting in release of a few 1000 terabecqureels of activity to a secondary containment where the material can be returned to a satisfactory storage area | More than a few per cent of the fuel gap (reactor coolant activity >300 uc/cc DEI) in a power reactor has been released into the reactor coolant from the fuel assemblies. | Release resulting in dose to the most exposed off-site individual on the order of .1 milliseverts (10mr)<br><br>Event results in worker exposure to 1 Gy (100R) | **Near accident with no safety layers remaining** - Events in which further failure of safety systems could lead to accident conditions or a situation in which safety systems would be inadequate if another event occurred | |
| 2 | INCIDENT | | **Significant Spread of contamination -** Release results in the presense of significant quantities of radioativity in the installation in areas not expected by design and which require corrective action | | Event results in dose rates in operating plant areas of > 50 mSv/hr (5 R/hr)<br><br>Event results in worker exposure exceeding a statutory annual dose limit. | **Incidents with significant failure in safety provisions -** but with sufficient defense in depth remaining to cope with additional failures | Add example: DEI or E elevated out of normal operating limits - requiring shutdown |
| 1 | ANOMALY | | | | | Event that results in operation beyond authorized operating limits and conditions | Add example: DEI or E elevated out of normal operating limits - but returned to within normal operating limits within specified action statement time limits |
| 0 Below Scale | DEVIATION | **No Safety Significance** | | | | **Anomaly beyond the authorized operating regime -** with operation remaining within authorized operating limits and conditions | Add example: DEI or E elevated but within normal operating limits - no MANDATORY LCO exists |
| Out of Scale | | No Safety Relevance | | | | | |

| US Classification Scale | | NUREG 0654/FEMA REP-1 | NEI 99-01, Rev 4 |
|---|---|---|---|
| GENERAL | Events are in process or have occurred which involve actual or imminent substantial core degradation or melting with potential for loss of containment integrity.<br><br>Releases can exceed EPA PAG exposure levels offsite | Loss of 2 of 3 fission product barriers with a potential loss of 3rd barrier.<br><br>Actual release projected dose rates offsite exceed EPA PAGs | Loss of 2 of 3 fission product barriers with a potential loss of 3rd barrier. |
| SITE AREA | Events are in process or have occurred which involve actual or likely major failures of plant functions needed for protection of the public<br><br>Releases not expected to exceed EPA PAGs except near site boundary | Degraded core with possible loss of coolable geometry | Loss or potential loss of Any two barriers |
| ALERT | Events are in process or have occurred which involve an actual or potential substantial degradation of the level of safety of the plant. Releases expected to be limited to a small fraction of the EPA PAGs. | Severe loss of fuel cladding<br>• High offgas at BWR air ejector monitor (>5Ci/sec)<br>• RCS activity >300 uC/cc DEI-131<br>• Failed fuel monitor increase > 1% fuel failure within 30 minutes or 5% total fuel failures | Any loss or potential loss of either the fuel clad or RCS barriers<br>• Core Cooling CSF orange or Red or CETC equivalent values exceeded<br>• Containment radiation (drywell for BWR) reading or RCS sample activity value indicates activity > 300 uc/cc DEI-131 (< 5% fuel clad damage)<br>• RVLIS indicates core uncovered |
| NOUE | Unusual events are in process or have occurred which indicate a potential degradation of the level of safety of the plant. No releases requiring offsite response are expected unless further degradation of safety systems occurs | Fuel damage indicated by:<br>• Hi offgas at BWR air ejector monitor (> 5e5 uc/sec)<br>• RCS sample indicates DEI spike > Tech Specs<br>• Failed fuel monitor indicates > 0.1 % equivalent fuel failure within 30 minutes | Fuel Clad degradation (SU4)(CU5) > Tech Spec allowable limits as indicated by rad monitor readings or coolant sample values. |
| 50.72 and 50.73 Reports | | • Initiation of S/D required by Tech Specs<br>• Plant in unanalyzed condition<br>• Plant in condition outside design basis<br>• Degraded spent fuel cask or confinement system | |

Draft Excerpt

REFERENCE

| Multipy number of | by | to obtain number of |
|---|---|---|
| becqurerels | 2.703 e-11 | cuires |
| nanocuries | 37 | becqurerels |
| curies | 3.7 e10 | becqurerels |
| sievert | 100 | rem |
| millisievert | 100 | millirem |
| rad | .01 | gray |
| gray | 100 | rad |

Source - NRC RTM-96 Table A-3 (PWR baseline coolant concentration)

- 4e-2 uc/cm3 I-131 in RCS Coolant normally (ANS-18.1)
- 2e4 uc/cm3 I-131 in RCS Coolant after gap release
- 1e5 uc/cm3 I-131 in RCS Coolant after in-vessel melt

Source - NRC RTM-96 Table A-4 (BWR baseline coolant concentration)

- 2e-3 uc/cm3 I-131 in RCS Coolant normally (ANS-18.1)
- 1e3 uc/cm3 I-131 in RCS Coolant after gap release
- 1e4 uc/cm3 I-131 in RCS Coolant after in-vessel melt

## 3.4 REACTOR COOLANT SYSTEM (RCS)

### 3.4.16 RCS Specific Activity

LCO 3.4.16 The specific activity of the reactor coolant shall be within limits.

APPLICABILITY: MODES 1 and 2,
MODE 3 with RCS average temperature ($T_{avg}$) ≥ 500°F.

ACTIONS

| CONDITION | REQUIRED ACTION | COMPLETION TIME |
|---|---|---|
| A. DOSE EQUIVALENT I-131 > 0.5 μCi/gm. | ----------Note----------<br>LCO 3.0.4 is not applicable.<br>------------------------------ | |
| | A.1 Verify DOSE EQUIVALENT I-131 within the acceptable region of Figure 3.4.16-1. | Once per 4 hours |
| | <u>AND</u> | |
| | A.2 Restore DOSE EQUIVALENT I-131 to within limit. | 48 hours |
| B. Gross specific activity of the reactor coolant not within limit. | B.1 Be in MODE 3 with $T_{avg}$ < 500°F. | 6 hours |

ACTIONS

| CONDITION | REQUIRED ACTION | COMPLETION TIME |
|---|---|---|
| C. Required Action and associated Completion Time of Condition A not met.<br><br>OR<br><br>DOSE EQUIVALENT I-131 in the unacceptable region of Figure 3.4.16-1. | C.1 Be in MODE 3 with $T_{avg} < 500°F$. | 6 hours |

SURVEILLANCE REQUIREMENTS

| SURVEILLANCE | | FREQUENCY |
|---|---|---|
| SR 3.4.16.1 | Verify reactor coolant gross specific activity $\leq 100/\bar{E}$ μCi/gm. | 7 days |
| SR 3.4.16.2 | -----NOTE-----<br>Only required to be performed in MODE 1.<br>----------<br><br>Verify reactor coolant DOSE EQUIVALENT I-131 specific activity ≤ 0.5 μCi/gm. | 14 days<br><br>AND<br><br>Between 2 and 6 hours after a THERMAL POWER change of ≥ 15% RTP within a 1 hour period |

SURVIELLANCE REQUIREMENTS

| SURVEILLANCE | | FREQUENCY |
|---|---|---|
| SR 3.4.16.3 | ----------------------------NOTE----------------------------<br>Not required to be performed until 31 days after a minimum of 2 effective full power days and 20 days of MODE 1 operation have elapsed since the reactor was last subcritical for ≥ 48 hours.<br>--------------------------------------------------------------<br><br>Determine Ē from a sample taken in MODE 1 after a minimum of 2 effective full power days and 20 days of MODE 1 operation have elapsed since the reactor was last subcritical for ≥ 48 hours. | 184 days |

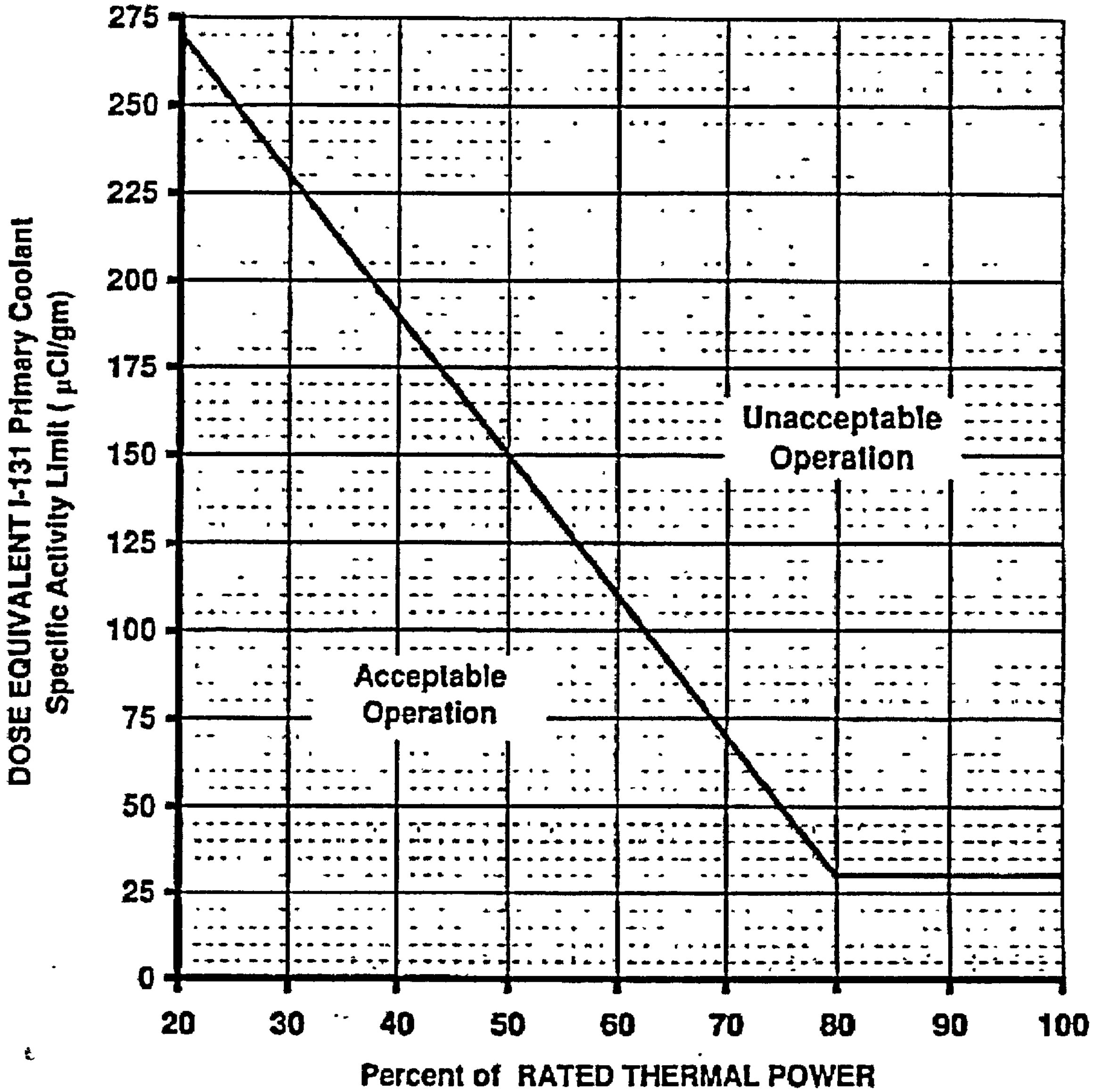



Figure 3.4.16-1

DOSE EQUIVALENT I-131 Primary Coolant Specific Activity Limit Versus Percent of RATED THERMAL POWER with the Primary Coolant Specific Activity > 0.5 µCi/gm DOSE EQUIVALENT I-131.

**MEETING WITH THE NUCLEAR ENERGY INSTITUTE (NEI) ON THE INTERNATIONAL NUCLEAR EVENT SCALE RATINGS**
**Attendance List**

July 26, 2002

| NAME | ORGANIZATION |
|---|---|
| ALAN NELSON | NUCLEAR ENERGY INSTITUTE |
| WALT LEE | SOUTHERN NUCLEAR |
| DEANN RALEIGH | LIS, SCIENTECH |
| GREGORY TWACHTMAN | MCGRAW-HILL |
| LANE HAY | SERCH BECHTEL |
| TERRY REIS | NRR/DRIP/OES |
| TOM BLOUNT | NRR/DIPM/EPHP |
| ROBERT STRANSKY | NSIR/DIRO/CS |
| JOELLE STAREFOS | RII/DRP/BROWNS FERRY |
| JERRY DOZIER | NRR/DRIP/OES |

Attachment 3